CHAPTER 1
INTRODUCTION AND SUMMARY

1.1 INTRODUCTION

[Historical Information] The Final Safety Analysis Report (FSAR) was submitted in support of an application by Consolidated Edison Company of New York, Inc. (Con Edison) for a license to operate the Indian Point Unit 2 nuclear power plant (AEC Docket 50-247, Permit No. CPPR-21). It provided pertinent technical information in accordance with Section 50.34 of 10 CFR 50 requirements to obtain a nuclear power plant operating license. Westinghouse Electric Corporation was the primary contractor and had turnkey responsibility for the design, construction, testing, and initial startup of the facility. Westinghouse had contracted with United Engineers and Constructors as architect-engineer to provide engineering assistance in the design of and construction of the structural and civil works.

This document is the Updated FSAR and is submitted in accordance with Section 50.71(e) of 10 CFR Part 50. The revision history is summarized in Section 1.9.2.

The following paragraphs contain a summary of the report's scope:

The unit employs a pressurized water reactor nuclear steam supply system (NSSS) furnished by Westinghouse Electric Corporation.

The reactor was originally licensed at a maximum thermal power of 2758 MW. By letter dated September 30, 1988 Con Edison initiated a request to authorize an increase in the licensed thermal power level to 3071.4 MW. The NRC documented their acceptance of this request in a Safety Evaluation Report dated March 7, 1990. By letter dated December 12, 2002, Entergy Nuclear Operations, Inc. initiated a request for a Measurement Uncertainty Recapture power uprate of 1.4%, for a licensed thermal power level of 3114.4 MW. The NRC documented their acceptance of this request in a Safety Evaluation Report dated May 22, 2003.

By letter dated January 29, 2004, Entergy Nuclear Operations, Inc. initiated a request for a Stretch Power Uprate, for licensed Thermal power level of 3216MW. The NRC documented their acceptance if this request in a Safety Evaluation Report dated October 27, 2004 and issued License Amendment #241.

[Deleted] The reactor power corresponds to an electric output from the turbine-generator of approximately 1078 MW.

The plant heat removal systems have been designed for the equivalent guarantee rating of 3071.4 MWt; some of the portions of the safety analysis dependent on heat removal capacity of plant and safeguards systems have assumed the maximum calculated power of 3216 MWt as have the evaluations of activity release and radiation exposure.

By NRC order dated August 27, 2001 (Reference 1), Con Edison's ownership/operation of Indian Point 1 and 2 was transferred to Entergy Nuclear Indian Point 2 (ENIP2), LLC, as the owner of Indian Point 1 and 2 plants, and Entergy Nuclear Operations (ENO), Inc. as the operator of Indian Point 2 and maintainer of Indian Point 1. Consequently, references to Con Edison (or derivatives thereof) in this document remain only when used in historical context.

The remainder of Chapter 1 of this report summarizes the principal design features and parameters of the plant, pointing out the similarities and differences with respect to other pressurized water nuclear power plants presently in operation. A general description of the plant is included as well as a statement and summary of all the General Design Criteria.

Chapter 2 contains a description and evaluation of the site and environs, supporting the suitability of the site for a reactor of the size and type described. Chapters 3 and 4 describe and evaluate the reactor and the reactor coolant system; Chapter 5, the containment system; Chapter 6, the engineered safety features; Chapter 7, plant instrumentation and control; Chapter 8, the electrical system; Chapter 9, the auxiliary and emergency system; Chapter 10, the steam and power conversion system; Chapter 11, radioactive waste disposal and radiation protection. Chapter 12 and 13 are conduct of operations and initial test and operations, respectively. They describe plant organization, training programs, and startup administrative procedure. Chapter 14 is a safety evaluation summarizing the analyses, which demonstrate the adequacy of the reactor protection system and the containment and engineered safety features, and which show that the consequences of various postulated accidents are within applicable limits.

REFERENCES FOR SECTION 1.1

1. NRC letter to Consolidated Edison, Indian Point Nuclear Generating Unit No.s 1 and 2 – Order Approving Transfer of Licenses from the Consolidated Edison Company of New York, Inc., to Entergy Nuclear Indian Point 2, LLC, and Entergy Nuclear Operations, Inc. and Approving Conforming Amendments (TAC Nos. MB0743 and MB0744), August 27, 2001.

1.2 SUMMARY PLANT DESCRIPTION

1.2.1 Site

Indian Point Unit 2 is adjacent to and north of Unit 1 on a site of approximately 239 acres of land on the east bank of the Hudson River at Indian Point, Village of Buchanan in upper Westchester County, New York. Indian Point Unit 3 (owned and operated by Entergy Nuclear and Entergy Nuclear Operations, Inc.) is adjacent to and south of Unit 1. The site is about 24 miles north of the New York City boundary line. The nearest city is Peekskill, 2.5 miles northeast of Indian Point. An aerial photograph, [Historical] Figure 2.2-1, shows the site and about 58 miles^2 of the surrounding area.

1.2.1.1 Meteorology

Meteorological conditions in the area of the site were determined during a 2-year test program. These data were used in evaluating the effects of gaseous discharges from the plant during normal operations and during the postulated loss-of-coolant accident. In addition, data supplied by the U.S. Weather Bureau at the Bear Mountain Station, regarding the meteorological conditions during periods of precipitation, were used to evaluate the rainout of fission gases into surface water reservoirs following the postulated loss-of-coolant accident. The evaluations indicate that the site meteorology provides adequate diffusion and dilution of any released gases.

1.2.1.2 Geology and Hydrology

Geologically, the site consists of a hard limestone in a jointed condition, which provides a solid bed for the plant foundation. The bedrock is sufficiently sound to support any loads, which could be anticipated up to 50 tons per ft^2, which is far in excess of any load, which may be imposed by the plant. Although it is hard, the jointed limestone formation is permeable to water. Thus, if water from the plant should enter the ground (an improbable event since the plant is designed to preclude any leakage into the ground) it would percolate to the river rather than enter any ground water supply. Additional studies by geology consultant, Thomas W. Fluhr, and examination of soil borings confirmed the above conclusions.

In the Hudson River, about 80,000,000 gallons of water flow past the plant each minute during the average tidal flow. This flow provides additional mixing and dilution for liquid discharges from the facility. In fact, however, this aspect is superfluous since the assumption in the plant design is to treat the river water as if it were used for drinking and thus to reduce radioactive discharges, by dilution with ordinary plant effluent, to concentrations that would be tolerable for drinking water. There is minimal danger of flooding at the site as discussed in Section 2.5.

1.2.1.3 Seismology

Seismic activity in the Indian Point area is limited to low-level microseismicity. Detailed field investigations (e.g., Ratcliffe, 1976, 1980; Dames and Moore, 1977) have been conducted in the immediate vicinity of Indian Point and along the major faults in the region. To date, no evidence has been found in the rocks exposed at the surface or sediments overlying fault traces or in cores obtained in the vicinity of Indian Point, that might support a conclusion that displacement has occurred along major fault systems within the New York Highlands, the Ramapo or its associated branches during Quaternary time (the last 1.5 million years). In the vicinity of Indian Point, evidence that no displacement has occurred in the last 65 million years (since the Mesozoic) along specific major structures has been observed.

The plant is designed to withstand an earthquake of Modified Mercalli Intensity VII. The validity of the selection of an Intensity VII earthquake was adjudicated before the Atomic Safety and Licensing Appeal Board. The Appeal Board's decision (ALAB-436) verified Intensity VII as the design basis earthquake for the plant.

1.2.1.4 Environmental Radiation Monitoring

Environmental radioactivity has been measured at the site and surrounding area for nearly 20 years in association with the operation of the three Indian Point Units. These measurements will be continued and reported. The radiation measurements of fallout, water samples, vegetation, marine life, etc., have shown no significant postoperative increase in activity. Noticeable increases in fallout have coincided with weapons-testing programs and appear to be related almost entirely to those programs. The New York State Department of Health in an independent 2-year postoperative study found that environmental radioactivity in the vicinity of the site is no higher than anywhere else in the State of New York.

1.2.1.5 Conclusions

Consideration of all the items mentioned above, plus the containment design and the engineered safety features included in the plant design lead to the conclusion of appropriate suitability of the site for safe operation of the Indian Point Unit 2 nuclear power plant. Accident analyses presented in Chapter 14 verify that the maximum expected doses at or beyond the site boundary are within applicable limits.

1.2.2 Plant Description

The unit incorporates a closed-cycle pressurized water nuclear steam supply system, a turbine generator and their necessary auxiliaries. A radioactive waste disposal system, fuel handling system and all auxiliaries, structures, and other onsite facilities required for a complete and operable nuclear power plant are provided for the unit.

The general arrangement of the plant is shown on historical Figures 1.2-1 and 1.2-4, and Figure 2.2-2. Other general plant arrangement drawings have been removed due to security reasons following September 11, 2001 and can be viewed as plant drawings 9321-2510, 9321-2511, 9321-2514, 9321-2517, 9321-3052, and 209812.

1.2.2.1 Nuclear Steam Supply System (NSSS)

The nuclear steam supply system consists of a pressurized water reactor, reactor coolant system, and associated auxiliary fluid systems. The reactor coolant system is arranged as four closed reactor coolant loops, each containing a reactor coolant pump and a steam generator, connected in parallel to the reactor vessel. An electrically-heated pressurizer is connected to one of the loops.

The reactor core is composed of uranium-dioxide pellets enclosed in zircaloy tubes with welded end plugs. The tubes are supported in assemblies by a spring clip grid structure. The mechanical control rods consist of clusters of stainless steel clad absorber rods and guide tubes located within the fuel assembly. The core was initially loaded in three regions of different enrichments with new fuel being introduced into the outer region at successive refuelings and discharged to spent fuel storage, following burnup.

The steam generators are vertical U-tube units employing Inconel tubes. Integral separating equipment reduces the moisture content of the steam leaving the steam generators to 0.25-percent or less.

The reactor coolant pumps are vertical, single stage, centrifugal pumps equipped with controlled leakage shaft seals.

Auxiliary systems are provided to perform the following functions:

1. Charge the reactor coolant system.
2. Add makeup water.
3. Purify reactor coolant water.
4. Provide chemicals for corrosion inhibition and reactor control.
5. Cool system components.
6. Remove residual heat when the reactor is shut down.
7. Cool the spent fuel storage pool.

8. Sample reactor coolant water.
9. Provide for emergency core cooling.
10. Collect reactor coolant drains.
11. Provide containment spray.
12. Provide containment ventilation and cooling.
13. Dispose of liquid, gaseous and solid wastes.

1.2.2.2 Reactor and Plant Control

The reactor is controlled by a coordinated combination of chemical shim and mechanical control rods. The control system allows the unit to accept step load changes of ± I0-percent and ramp load changes of ± 5-percent per min over the load range of 15-percent to, but not exceeding, 100-percent power under nominal operating conditions subject to xenon limitations.

1.2.2.3 Turbine and Auxiliaries

The turbine is a tandem-compound, comprising one high pressure and three low pressure cylinders, 1800 rpm unit having 45-in. exhaust blading in the low pressure cylinders. There are four moisture preseparators located at the four high-pressure turbine exhaust lines and six combination moisture separator-reheater units that are employed to dry and superheat the steam between the high and low pressure turbine cylinders. The turbine generator is capable of a 50-percent loss of external electrical load without turbine or reactor trip. The turbine auxiliaries include deaerating surface condensers, steam jet air ejector, turbine-driven main feedwater pumps, motor-driven condensate pumps, and six stages of feedwater heating.

The original turbine generator had a guaranteed capability of 1,021,793 kWe at 1.5-in. Hg absolute exhaust pressure with zero percent makeup and six stages of feedwater heating.

1.2.2.4 Electrical System

The main generator feeds electrical power through an isolated phase bus to two half-sized main power transformers. Station auxiliaries receive power during normal operation from either the station auxiliary transformer (i.e., offsite power) or the unit auxiliary transformer (i.e., unit main power transformers).

The auxiliary electrical system provides power to those auxiliary components required to operate during normal or emergency conditions of operation. Standby power required during plant startup, shutdown, and after reactor trip is supplied to the station auxiliary transformer from the Con Edison 138-kV system by either of two separate overhead lines from the Buchanan substation approximately 0.50 mile from the plant. Alternate feeds from the 13.8-kV system are also available for immediate manual connection to the auxiliary buses.

Emergency power supply for vital instruments and controls is from four 125-V station batteries.

The system design provides sufficient independence, isolation capability, and redundancy between the different power sources to avoid complete loss of auxiliary power.

1.2.2.5 Control Room

The plant is provided with a reactor and turbine-generator control room containing all necessary instrumentation for the operation of the plant under normal and accident conditions.

Adequate shielding and air conditioning facilities permit occupancy during all operating or accident conditions.

1.2.2.6 Diesel Generators

Three diesel-generator sets supply emergency power for shutdown or essential safeguards operation in the event of a loss of all other alternating current auxiliary power.

1.2.2.7 Waste Disposal System

The waste disposal system collects and processes liquids, gaseous, and solid waste from plant operation for removal from plant site. All removals are made in accordance with government guidelines for the process.

1.2.2.8 Fuel Handling System

The fuel handling system provides the ability to fuel and refuel the reactor core. Carefully established administrative procedures plus the design of the system minimizes the probability of potential fission product release during the refueling operation.

The system also includes the following features:

1. Safe accessibility for operating personnel.
2. Provisions to prevent fuel storage criticality.
3. Visual monitoring of the refueling procedures at all times.

1.2.2.9 Engineered Safety Features

The engineered safety features for this plant have sufficient redundancy of component and power sources such that under the conditions of a hypothetical loss-of-coolant, the system can, even when operating with partial effectiveness, maintain the integrity of the containment and keep the exposure of the public within applicable limits.

The major engineered safeguards systems are as follows:

1. The containment system, which incorporates continuously pressurized and monitored penetrations and liner weld channels and a seal water injection system, which provides a highly reliable, essentially leaktight barrier against the escape of radioactivity, which might be released to the containment atmosphere.

2. The safety injection system (which constitutes the emergency core cooling system) provides borated water to cool the core in the event of a loss-of-coolant accident.

3. The containment air recirculation cooling system provides a heat sink to cool the containment atmosphere.

4. The containment spray system provides a spray of cool, borated water to the containment atmosphere that is a heat sink and also provides iodine removal capability.

1.2.2.10 Structures

The major structures are the reactor containment building, the primary auxiliary building, the control building, the fuel storage building, the turbine building, and the maintenance and operations building. General layouts of the reactor containment interior components arrangement are shown on Plant Drawings 9321-2501, 9321-2502, 9321-2503, 9321-2506, 9321-2507, 9321-2508 [formerly UFSAR Figures 5.1-2 through 5.2-7]. General layouts and interior components arrangement of the primary auxiliary building, control building, fuel storage building, and holdup tank building were removed from the UFSAR due to security reasons following September 11, 2001 and can be viewed on plant drawings.

1.2.2.11 Containment

The reactor containment is a steel-lined reinforced concrete cylinder with a hemispherical dome and a flat base. The containment is designed to withstand the internal pressure accompanying a loss-of-coolant accident, is virtually leaktight, and provides adequate radiation shielding for both normal operation and accident conditions.

When required, the containment isolation valve seal water system permits automatic rapid sealing of pipes, which penetrate the containment so that in the event of any loss-of-coolant accident, leakage from containment to the environment is minimal.

Ground accelerations for the operational basis earthquake used for containment design purposes and all seismic Class I structures (Section 1.11) are 0.10g applied horizontally and 0.05g applied vertically. In addition, ground accelerations for the design basis earthquake of 0.15g horizontal and 0.10g vertical are used to analyze the no loss-of-function concept.

1.2 FIGURES

Figure No.	**Title**
Figure 1.2-1	Indian Point Nuclear Generating Units 1 & 2 [Historical]
Figure 1.2-2	Deleted
Figure 1.2-3	Deleted
Figure 1.2-4	Cross Section of Plant [Historical]
Figure 1.2-5	Deleted
Figure 1.2-6	Deleted
Figure 1.2-7	Deleted
Figure 1.2-7	Deleted
Figure 1.2-8	Deleted
Figure 1.2-9	Deleted

1.3 GENERAL DESIGN CRITERIA (GDC)

The General Design Criteria define or describe safety objectives and approaches incorporated in the design of this plant. These General Design Criteria, tabulated explicitly in the pertinent systems sections in this report, comprised the proposed Atomic Industrial Forum versions of the criteria issued for comment by the AEC on July 11, 1967. Also included in this section, are brief descriptions of related plant features, which are provided to meet the design objectives reflected in the criteria at the time of the initial license application. The descriptions are more fully developed in those succeeding sections of the report indicated by the references.

More recently, Con Edison completed a study of compliance with 10 CFR Parts 20 and 50 in accordance with the Commission's Confirmatory Order of February 11, 1980. The detailed results of the evaluation of Indian Point Unit 2 compliance with the then current General Design Criteria established by the Nuclear Regulatory Commission (NRC) in 10 CFR 50 Appendix A, were submitted to the NRC by Con Edison on August 11, 1980 (Reference 1). Commission concurrence was received on January 19, 1982.

The parenthetical numbers following the section headings indicate the numbers of their related proposed Atomic Industrial Forum versions of the General Design Criteria as described in the first paragraph of this section.

1.3.1 Overall Plant REQUIREMENTS (GDC 1 - GDC 5)

All systems and components of the facility are classified according to their importance. Those items vital to safe shutdown and isolation of the reactor or whose failure might cause or increase the severity of a loss-of-coolant accident or result in an uncontrolled release of excessive amounts of radioactivity are designated Class I. Those items important to reactor operation but not essential to safe shutdown and isolation of the reactor or control of the release of substantial amounts of radioactivity are designated Class II. Those items not related to reactor operation or safety are designated Class III.

Class I systems and components are essential to the protection of the health and safety of the public. Consequently, they are designed, fabricated, inspected, erected, and use materials selected to the applicable provisions of recognized codes, good nuclear practice and to quality standards that reflect their importance.

All systems and components designated Class I are designed so that there is no loss of function in the event of the maximum potential ground acceleration acting in the horizontal and vertical directions simultaneously. The working stresses of both Class I and Class II items are kept within code allowable values for the operational basis earthquake. Similarly, measures are taken in the plant design to protect against high winds, sudden barometric pressure changes, flooding, and other natural phenomena.

Reference sections:

Section Title	Section
Site & Environment;	
Meteorology	2.6
Geology and Seismology	2.7
Reactor Coolant System;	
Design Bases	4.1

Containment System Structures;	
Design Bases	5.1.1
Electrical Systems;	
Design Bases	8.1
Introduction & Summary;	
Design Criteria for Structures and Equipment	1.11

Fire prevention in all areas of the nuclear electric plant is provided by structure and component design, which maximizes the use of fire-resistant materials, optimizes the containment of combustible materials and maintains exposed combustible materials below their ignition temperature in the design atmosphere. Fixed and portable fire fighting equipment is provided with capacities proportional to the energy that might credibly be released by fire.

Reference sections:

Section Title	Section
Instrumentation & Control;	
Operating Control Stations	7.7
Auxiliary & Emergency Systems;	
Facility Service System	9.6

The only structures, systems, or components important to safety that are shared between Units 2 and 3 are:

1. The cooling water discharge channel, which carries the service water discharge to the river.
2. The Emergency Fuel Supply to the Emergency Diesel Generators.

Since the channel is designed to handle the discharge flow from both operating units, sharing of this structure will not impair the ability of Unit 2 safety systems to perform their safety functions. Units 2 and 3 are required by Technical Specification to maintain a designated amount of fuel reserve onsite or at the Buchanan substation, which is dedicated for emergency diesel generator use at that unit. Therefore sharing the Emergency Fuel Supply to the Emergency Diesel Generators will not impair the ability of Unit 2 safety systems to perform their safety functions.

Reference sections:

Section Title	Section
Electrical Systems;	
Emergency Fuel Supply	8.2.3.2
Auxiliary & Emergency Systems;	
Service Water System	9.6.1

A complete set of facility plant and system diagrams including arrangements, plans, and structural plans and records of initial tests and operation are maintained throughout the life of the reactor. A set of all the quality assurance data generated during fabrication and erection of the essential components of the plant, as defined by the quality assurance program, is retained.

Reference sections:

Section Title	Section
Conduct of Operations;	
Records	12.4

Initial Tests and Operation	13
Introduction & Summary;	
Quality Assurance Program	1.10

1.3.2 Protection By Multiple Fission Product Barriers (GDC 6 - GDC 10)

The reactor core, with its related control and protection system, is designed to function throughout its design lifetime without exceeding acceptable fuel damage limits. The core design, together with reliable process and decay heat removal systems, provides for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and anticipated transient situations.

The reactor control and protection instrumentation is designed to actuate a reactor trip for any anticipated combination of plant conditions when necessary to ensure DNBR remains at or above the applicable safety analysis DNBR limit and fuel center temperature below the melting point of uranium-dioxide.

Reference sections:

Section Title	Section
Reactor;	
Design Bases,	3.1
Reactor Design	3.2
Instrumentation and Control;	
Protective Systems	7.2
Safety Analysis	14

The design of the reactor core and related protection systems ensures that power oscillations, which could cause fuel damage in excess of acceptable limits are not possible or can be readily suppressed.

Low frequency spatial xenon oscillations may occur in the axial dimension. However, the core is expected to be (and has proven to be) stable to xenon oscillations in the X-Y dimension. Ex-core instrumentation is provided to monitor any xenon induced oscillations. Incore instrumentation is used periodically to calibrate and verify the information provided by the ex-core instrumentation.

The moderator temperature and overall power coefficients in the power operating range were maintained negative by inclusion of burnable poison shims in the first core loading. The overall power coefficient in the power operating range is always negative (as discussed in Section 14.1.11.2).

Reference sections:

Section Title	Section
Reactor;	
Design Bases,	3.1
Reactor Design	3.2

The reactor coolant system in conjunction with its control and protective provisions is designed to accommodate the system pressures and temperatures attained under all expected modes of plant operation or anticipated system interactions, and maintain the stresses within applicable code stress limits. The materials of construction of the pressure retaining boundary of the reactor coolant system are protected by control of coolant chemistry from corrosion phenomena, which might otherwise reduce the system structural integrity during its service lifetime.

System conditions resulting from anticipated transients or malfunctions are monitored and appropriate action is initiated automatically to maintain the required cooling capability and to limit system conditions so that continued safe operation is achieved.

The system is protected from overpressure by means of pressure relieving devices, as required by Section III of the ASME Boiler and Pressure Vessel Code and by an overpressure protection system intended to ensure compliance with 10 CFR 50, Appendix G.

Isolable sections of the system are provided with overpressure relieving devices discharging to closed systems such that the system code allowable relief pressure within the protected section is not exceeded.

Reference section:

Section Title	Section
Reactor Coolant System;	
Design Bases	4.1

The design pressure of the containment exceeds the peak pressure occurring as the result of the complete blowdown of the reactor coolant through any pipe rupture of the reactor coolant system up to and including the hypothetical severance of a reactor coolant system pipe.

All piping systems, which penetrate the vapor barrier are anchored so that the penetration is structurally adequate to resist the piping loads and the vapor barrier will not be breached due to a hypothesized pipe rupture. The lines (with the exception of sample tubing) connected to the reactor coolant system that penetrate the vapor barrier are restrained near the secondary shield walls and are each provided with at least one valve between the shield wall and the reactor coolant system. These restraints are designed to withstand the thrust moment and torque resulting from a hypothesized rupture of the attached pipe.

All isolation valves are supported to withstand, without impairment of valve operability, the combined loading of the design-basis accident and design seismic conditions.

Reference section:

Section Title	Section
Containment System Structures;	
Design Bases	5.1.1

1.3.3 Nuclear And Radiation Controls (GDC 11 - GDC 18)

The plant is equipped with a control room, which contains all controls and instrumentation and facilities necessary for continuous operation of the reactor and turbine generator under normal and accident conditions.

Sufficient shielding, ventilation control and filtration, and containment integrity are provided to ensure that control room personnel will not be subjected to doses under postulated accident conditions during occupancy of the control room, which in the aggregate, would exceed the applicable limits.

Instrumentation and controls essential to avoid undue risk to the health and safety of the public are provided to monitor and maintain neutron flux, reactor coolant pressure, flow rate, temperature, and control rod positions within prescribed operating ranges.

The non-nuclear regulating, process and containment instrumentation measures temperatures, pressures, flows, and levels in the reactor coolant system, steam systems, containment, and other auxiliary systems.

The quantity and types of process instrumentation provided ensures safe and orderly operation of all systems and processes over the full operating range of the plant.

The operational status of the reactor is monitored from the control room. When the reactor is subcritical, the neutron source multiplication is continuously monitored and indicated by proportional counters located in instrument wells in the primary shield adjacent to the reactor vessel. The source detector channels are checked prior to operations in which criticality may be approached by the use of an incore source. Any appreciable increase in the neutron source multiplication, including that caused by the maximum physical boron dilution rate, is slow enough to give ample time to start corrective action (boron dilution stop and/or emergency boron injection) to prevent the core from becoming critical (as discussed in Sections 14.1.5.2.3 and 14.1.5.3).

When the reactor is critical, means for showing the relative reactivity status of the reactor is provided by display of all rod bank positions in the control room. Periodic samples of the coolant boron concentration are taken, the variation of which provides a further check on the reactivity status of the reactor including core depletion during life.

Instrumentation and controls provided for the protective systems are designed to trip the reactor when necessary to prevent or limit fission product release from the core, to limit energy release, to signal closure of containment isolation valves, and to control the operation of engineered safety features equipment.

During reactor operation in the startup and power modes, redundant safety limit signals will automatically actuate two reactor trip breakers, which are in series with the rod drive mechanism coils. This action would interrupt power and initiate reactor trip.

Reference sections:

Section Title	Section
Instrumentation and Control;	
General Design Criteria,	7.1
Protective Systems,	7.2
Nuclear Instrumentation,	7.4
Operating Control Stations	7.7

The reactor protection system receives signals from plant instrumentation, which are indicative of an approach to an unsafe operating condition, actuates alarms, prevents control rod motion, initiates load cutback, and/or opens the reactor trip breakers, depending on the severity of the condition.

The basic reactor tripping philosophy is to define a region of power and coolant temperature conditions allowed by the primary tripping functions (e.g., overpower ΔT trip, overtemperature ΔT trip, and nuclear flux trips). The allowable operating region within these trip settings is shown to prevent any combination of power, temperature, and pressure, which would result in DNB with all reactor coolant pumps in operation. Additional tripping functions such as low and high pressurizer pressure trips, high pressurizer level trip, loss of flow trip, steam and feedwater flow mismatch trip, steam generator low-low level trip, turbine trip, safety injection trip, nuclear flux trips (source, intermediate, and high range), and manual trip are provided as backup to the primary tripping functions for specific accident conditions and mechanical failures.

Intermediate Range and Power Range rod stops, Overtemperature ΔT and Overpower ΔT rod stops, are provided to prevent abnormal power conditions which could result from excessive control rod withdrawal initiated by operator violation of administrative procedures.

Reference sections:

Section Title	Section
Engineered Safety Features;	
Safety Injection System	6.2
Instrumentation and Control;	
Protective Systems	7.2

Positive indications in the control room of leakage of coolant from the reactor coolant system to the containment are provided by equipment, which permits continuous monitoring of containment air activity and humidity, and of runoff from the condensate collecting pans under the cooling coils of the containment air recirculation units. The basic design criterion is the detection of deviations from normal containment environmental conditions including air particulate activity, radiogas activity, humidity, condensate runoff and in addition, in the case of gross leakage, the liquid inventory in the process systems and containment sump.

The containment atmosphere, the plant vent, the containment fan-coolers service water discharge, the waste disposal system liquid effluent, and the component cooling loop are monitored for radioactivity concentration during all normal operations, anticipated transients, and accident conditions.

For the case of leakage from the reactor containment under accident conditions, the plant area radiation monitoring system supplemented by portable survey equipment to be kept in the control room provide adequate monitoring of accident releases.

Monitoring and alarm instrumentation is provided for fuel and waste storage and handling areas to detect excessive radiation levels. The permanent record of activity releases is provided by radiochemical analysis of known quantities of waste.

A controlled ventilation system removes gaseous radioactivity from various areas of the plant and discharges it to the atmosphere via the plant vent. Radiation monitors are in continuous service in these areas to actuate high activity alarms on the control board annunciator and initiate containment isolation.

Reference sections:

Section Title	Section
Engineered Safety Features;	
Safety Injection System,	6.2
Isolation Valve Seal Water System,	6.5
Leakage Detection and Provisions for the Primary and Auxiliary Coolant Loops	6.7
Auxiliary & Emergency Systems;	
Auxiliary Coolant System	9.3
Waste Disposal & Radiation Protections System;	
Radiation Protection	11.2

1.3.4 Reliability And Testability Of Protection Systems (GDC 19 - GDC 26)

Upon a loss of power to the magnetic-type control rod drive mechanisms, the rod cluster control (RCC) assembly is released and falls by gravity into the core. The reactor internals, fuel assemblies, RCC assemblies and drive system components are designed as Class I equipment. The RCC assemblies are fully guided through the fuel assembly and for the maximum travel of the control rod into the guide tube. Furthermore, the RCC assemblies are never fully withdrawn from their guide thimbles in the fuel assembly. Due to this and the flexibility designed into the RCC assemblies, abnormal loadings and misalignments can be sustained without impairing operation of the RCC assemblies.

All reactor protection channels are supplied with sufficient redundancy to provide the capability for channel calibration and test at power. The analog channel trip bistables have the capability to be bypassed for surveillance testing. This “test in bypass” feature enables analog logic relays in a protective channel to be bypassed while testing actuation of the associated bistable. Bistable testing does not preclude the protective action provided by concurrent channels.

Two reactor trip breakers are provided to interrupt power to the rod drive mechanisms. The breaker main contacts are connected in series with the power supply to the mechanism coils. Opening either breaker interrupts power to all full length rod mechanisms. Each breaker is opened through an undervoltage trip coil. Each protection channel actuates two separate trip logic trains, one for each reactor trip breaker undervoltage trip coil. The protection system is thus inherently safe in the event of a loss of rod control power.

Channel independence is carried throughout the system extending from the sensor to the relay providing the logic. In most cases, the safety and control functions when combined are combined only at the sensor. A failure in the control circuitry does not affect the safety channel. This approach is used for pressurizer pressure and water level channels, steam generator water level, T_{avg} and ΔT channels, steam flow, and nuclear power range channels. The power supplied to the channels is fed from four vital instrument buses. All four of the buses are supplied by static inverters.

The initiation of the engineered safety features provided for loss-of-coolant accidents (e.g., high head safety injection, residual heat removal pumps, and containment spray systems) is accomplished from redundant signals derived from reactor coolant system and containment instrumentation. The initiation signal for containment spray comes from coincidence of two sets of two-out-of-three high-high containment pressure signals. On loss of voltage to the safety features equipment buses, the emergency diesel generators will be automatically started and connected to their respective buses.

Trip signals for the containment isolation valves are derived from either a high-high containment pressure signal (phase B), and/or a safety injection signal (phase A), and/or a containment ventilation isolation signal.

The components of the protection system are designed and laid out so that the mechanical and thermal environment accompanying any emergency situation in which the components are required to function does not interfere with that function.

Each protection channel in service at power is capable of being calibrated and tripped independently by simulated signals for test purposes to verify its operation.

Redundancy is provided in that there are three emergency diesel-generator sets capable of supplying separate 480-V buses. The minimum complement of safety features equipment is supplied from any two of the three emergency diesel generators.

The ability of the emergency diesel-generator sets to start within the prescribed time and to carry load is periodically checked.

An open circuit or loss of reactor trip channel power causes the system to go into its trip mode. In a two-out-of-three circuit, the three channels are equipped with separate primary sensors and each channel is energized from independent electrical buses.

The signal for containment isolation is developed from a two-out-of-three circuit in which each channel is separate and independent and which signals for containment isolation upon loss of power. The failure of any channel to de-energize when required does not interfere with the proper functioning of the isolation circuit.

Diesel engine cranking is accomplished by a stored energy system supplied solely for the associated emergency diesel generator. The undervoltage relay scheme is designed so that loss of 480-V power does not prevent the relay scheme from functioning properly.

Reference sections:

Section Title	Section
Instrumentation and Control; Protection Systems	7.2
Electrical Systems	8

1.3.5 Reactivity Control (GDC 27 - GDC 32)

Reactivity control is achieved by two independent systems, the rod cluster control assemblies and the chemical and volume control system, which regulates the concentration of boric acid solution neutron absorber in the reactor coolant system. The system is designed to prevent, under anticipated system malfunction, uncontrolled or inadvertent reactivity changes, which might stress the system beyond design limits.

The reactivity control systems provided are capable of making and holding the core subcritical from any hot standby or hot operating condition, including those resulting from power changes.

The maximum excess reactivity expected for the core occurs for the cold, clean condition at the beginning-of-life (as discussed in Section 14.1.11.2).

The full length rod cluster control assemblies are divided into two categories comprising control banks and shutdown banks. The control bank of the rod cluster control assemblies is used to compensate for short-term reactivity changes at power produced due to variations in reactor power or in coolant temperature. The chemical shim control is used to compensate for the more slowly occurring changes in reactivity throughout core life such as those due to fuel depletion and fission product buildup and decay.

The shutdown banks are provided to supplement the control banks of the rod cluster control assemblies to make the reactor at least 1-percent subcritical (^{k}eff = 0.99) following trip from any credible operating condition to the hot, zero power condition assuming the most reactive rod cluster control assembly remains in the fully withdrawn position.

Boron injection from the safety injection system supplements rod insertion and prevents exceeding core safety limits in the event of the maximum credible steam break, namely opening of a safety valve. This is accomplished with maximum worth rod fully withdrawn.

Any time that the plant is at power, the quantity of boric acid retained in the boric acid storage tanks and ready for injection always exceeds that quantity required for the normal cold shutdown.

The boric acid solution is transferred from the boric acid storage tanks into the reactor coolant by the boric acid transfer pumps and charging pumps, which can be operated from emergency diesel-generator power on loss of primary power. Boric acid can be injected by either boric acid transfer pumps and one of the three charging pumps to shut down the reactor from full power with no rods inserted. In addition, boric acid can be injected to compensate for xenon decay (xenon decay below the equilibrium operating level will not actually begin until approximately 20 hr after shutdown). Additional boric acid is employed if it is desired to bring the reactor to cold shutdown conditions.

The reactor protection systems will limit reactivity transients such that DNBR remains at or above the applicable safety analysis DNBR limit due to any single malfunction in the reactor coolant deboration controls.

Limits, which include considerable margin, are placed on the maximum reactivity worth of control rods by limiting position of insertion as a function of power and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals so as to lose capability to cool the core.

The rod cluster drive mechanisms are wired into preselected groups and are prevented from being withdrawn in other than their respective groups. The rod drive mechanism is of the magnetic latch type and the coil actuation is sequenced to provide variable speed rod travel. The maximum reactivity insertion rate is analyzed in the detailed plant analysis assuming two of the highest worth banks to be accidentally withdrawn at maximum speed, yielding reactivity insertion rates of the order of 8×10^{-4} Δk/sec, which is well within the capability of the overpower-overtemperature protection circuits to prevent core damage (as discussed in Section 14.1.1).

Reference sections:

Section Title	Section
Reactor;	
Design Bases	3.1
Instrumentation and Control;	
Protective Systems,	7.2
Regulating Systems	7.3
Auxiliary & Emergency Systems;	
Chemical and Volume Control System	9.2

1.3.6 Reactor Coolant Pressure Boundary (GDC 33 - GDC 36)

The reactor coolant pressure boundary is shown to be capable of accommodating without further rupture, the static and dynamic loads imposed as a result of a sudden reactivity insertion such as rod ejection (as discussed in Section 14.2.6.10).

The operation of the reactor is such that the severity of an ejection accident is inherently limited. Since the rod cluster control assemblies are used to control load variations only and core depletion is followed with boron dilution, only the rod cluster control assemblies in the controlling groups are inserted in the core at power, and at full power these rods are only partially inserted. A rod insertion limit monitor is provided as an administrative aid to the operator to ensure that this condition is met.

By using the flexibility in the selection of control rod groupings, radial locations and positions as a function of load, the design limits the maximum fuel temperature for the highest worth ejected rod to a value, which precludes any resultant damage to the primary system pressure boundary.

The failure of a rod mechanism housing causing a rod cluster control assembly to be rapidly ejected from the core is evaluated as a theoretical, though not a credible, accident. The analysis is discussed in Section 14.2.6.

In the core region of the reactor vessel, the V-notch toughness of the material will change during operation as a result of fast neutron exposure, which results in a shift in the nil ductility transition temperature (NDTT). This is factored into the operating procedures in such a manner that full operating pressure is not obtained until the affected vessel material is above the increased design transition temperature (DTT) and in the ductile material region. The pressure during startup and shutdown at the temperature below NDTT is maintained below the threshold of concern for safe operation.

The DTT is a minimum NDTT plus 60^{o}F and dictates the procedures to be followed in the hydrostatic test and in station operations to avoid excessive cold stress. The value of the DTT is increased during the life of the plant as required by the expected shift in the NDTT, and as confirmed by the experimental data obtained from irradiated specimens of reactor vessel materials during the plant lifetime.

The design of the reactor vessel and its arrangement in the system provide the capability for accessibility during service life to the entire internal surfaces of the vessel and the external zones of the vessel including the nozzle to reactor coolant piping welds and the top and bottom heads. The reactor arrangement within the containment provides sufficient space for inspection of the external surfaces of the reactor coolant piping, except for the area of piping within the primary shielding concrete.

Determination of the NDTT of the core region plate forgings, weldments, and associated heat treated zones is performed in accordance with ASTM E-185, Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors. Samples of reactor vessel plate material have been retained and catalogued in case future engineering development shows the need for further testing.

The material properties surveillance program includes not only the conventional tensile and impact tests, but also fracture mechanics specimens. The observed shifts in NDTT of the core region materials with irradiation will be used to confirm the calculated limits to startup and shutdown transients.

To define permissible operating conditions below DTT, a pressure range is established, which is bounded by a lower limit for pump operation and an upper limit, which satisfies reactor vessel stress criteria. Since the normal operating temperature of the reactor vessel is well above the maximum expected DTT, brittle fracture during normal operation is not considered to be a credible mode of failure.

Reference sections:

Section Title	Section
Reactor Coolant System;	
System Design and Operation,	4.2
Safety Limits and Conditions,	4.4
Inspections and Tests,	4.5
Determination of Reactor Pressure Vessel NDTT	Appendix 4A

1.3.7 Engineered Safety Features (GDC 37 - GDC 65)

The design, fabrication, testing and inspection of the core, reactor coolant pressure boundary, and their protection systems give assurance of safe and reliable operation under all anticipated normal, transient, and accident conditions. However, engineered safety features are provided in the facility to back up the safety provided by these components.

These engineered safety features have been designed to cope with any size reactor coolant pipe break up to and including the circumferential rupture of any pipe in that boundary assuming unobstructed discharge from both ends as discussed in Section 14.3.3.3. They are also designed to cope with any steam or feedwater line break up to and including the main steam or feedwater headers as discussed in Section 14.2.5. The total loss of all offsite power is assumed concurrent with these accidents.

The release of fission products from the reactor fuel is limited by the safety injection system, which by cooling the core and limiting the fuel clad temperature, keeps the fuel in place and substantially intact with its essential heat transfer geometry preserved and limits the metal-water reaction to an insignificant amount.

The basic design criteria for ensuring that the core geometry remains in place and substantially intact so that effective cooling of the core is not impaired following a loss-of-coolant accident:

1. The cladding temperature is to be less than:
    a. The melting temperature of Zircaloy-4
    b. The temperature at which gross core geometry distortion, including fragmentation, may be expected.

2. The total core metal-water reaction will be limited to less than 1-percent.

The safety injection system (which constitutes the emergency core cooling system) consists of high and low head centrifugal pumps driven by electric motors, and passive accumulator tanks, which are self-energized and which act independently of any actuation signal or power source.

The release of fission products from the containment is limited in three ways:

1. Blocking the potential leakage paths from the containment. This is accomplished by:
    a. A steel-lined concrete reactor containment with continuously pressurized double-sealed penetrations and liner weld channels, which form a virtually leaktight barrier to the escape of fission products should a loss-of-coolant accident occur.

    b. Isolation of process lines by the containment isolation system, which imposes double barriers in each line penetrating the containment.

2. Reducing the fission product concentration in the containment atmosphere. This is accomplished by containment spray, which removes elemental iodine vapor and particulates from the containment atmosphere by washing action.

3. Reducing the containment pressure and thereby limiting the driving potential for fission product leakage. This is accomplished by cooling the containment atmosphere by the following independent systems, each with adequate heat removal capacity:
    a. Containment spray system
    b. Containment air recirculation cooling system.

A comprehensive program of plant testing is performed for all equipment systems and system control vital to the functioning of engineered safety features. The program consists of performance tests of individual pieces of equipment in the manufacturer's shop and integrated tests of the system as a whole, and periodic tests of the actuation circuitry and mechanical components to assure reliable performance, upon demand, throughout the plant lifetime. In the event that one of the components should require maintenance as a result of failure to perform during the test according to prescribed limits, the necessary corrections or minor maintenance will be made and the unit retested immediately.

The plant is supplied with emergency power sources as follows:

1. Three independent emergency diesel generators, located in the Diesel Generator Building adjacent to the Primary Auxiliary Building, supply emergency power to the engineered safety features buses in the event of a loss of AC auxiliary power. There are no automatic bus ties associated with these buses. Each diesel generator is started automatically on a safety injection signal or upon the occurrence of an undervoltage

condition on any vital 480-V switchgear bus. The system is sufficiently redundant such that any two diesels have adequate capacity to supply the engineered safety features for the design basis accident concurrent with a loss of offsite power. One diesel is adequate to provide power for a safe and orderly plant shutdown in the event of a loss-of-offsite electrical power.

2. Emergency power for vital instrumentation and control and for emergency lighting is supplied from the 125 VDC system via four independent DC channels. The station batteries supply emergency power to the instrumentation and control systems when their associated battery chargers are not available.

For such engineered safety features as are required to ensure safety in the event of an accident, protection from dynamic effects or missiles is considered in the layout of plant equipment and missile barriers.

Layout and structural design specifically protect injection paths leading to unbroken reactor coolant loops against damage as a result of the maximum reactor coolant pipe rupture. Injection lines penetrate the main missile barrier, and the injection headers are located in the missile-protected area between the missile barrier and the containment outside wall. Individual injection lines, separated to the maximum extent practicable, are connected to the injection header, pass through the barrier and then connect to the loops. Movement of the injection line, associated with rupture of a reactor coolant loop, is accommodated by line flexibility and by the design of the pipe supports such that no damage outside the missile barrier is possible.

In 1989, the NRC approved changes to the design basis with respect to dynamic effects of postulated primary loop ruptures, as discussed in Section 4.1.2.4.

Each engineered safety feature provides sufficient performance capability to accommodate any single failure of an active component and still function in a manner to avoid undue risk to the health and safety of the public.

Under the hypothetical accident conditions, the containment air recirculation cooling system, and the containment spray system are designed and sized so that either system, operating alone at its rated capacity, is able to supply the necessary postaccident cooling capacity to reduce rapidly the containment pressure following blowdown and cooling of the core by safety injection. Together these two systems provide the single failure protection for the containment cooling function as analyzed in Chapter 14.

All active components of the safety injection system (exception: injection line isolation valves) and the containment spray system are located outside the containment and not subject to containment accident conditions.

Instrumentation, motors, cables, and penetrations located inside the containment are selected to meet the most adverse accident conditions to which they may be subjected. These items are either protected from containment accident conditions or are designed to withstand, without failure, exposure to the worst combination of temperature, pressure, radiation, and humidity expected during the required operational period.

The reactor is maintained subcritical following a primary system pipe rupture accident. Introduction of borated cooling water into the core results in a net negative reactivity addition (as discussed in Section 14.3.2). The control rods insert and remain inserted, except for the large break loss of coolant accident analysis where it is conservatively assumed that the control rods do not insert as discussed in Section 14.3.3.2. The delivery of cold safety injection water to the reactor vessel following accidental expulsion of reactor coolant was evaluated to ensure that this does not cause further loss of integrity of the reactor coolant system boundary (as discussed in Section 14.3.4.3.3).

Design provisions have been made to the extent practical to facilitate access to the critical parts of the reactor vessel internals, injection nozzles, pipes, valves, and safety injection pumps for visual or boroscopic inspection for erosion, corrosion, and vibration wear evidence, and for non-destructive test inspection where such techniques are desirable and appropriate.

Design provisions are made so that active components of the safety injection system can be tested periodically for operability and functional performance. The safety injection pumps can be tested periodically during plant operation using the minimum flow recirculation lines provided. The residual heat removal pumps are used every time the residual heat removal loop is put into operation.

An integrated safety injection system test is performed at refueling outage intervals. This test does not introduce flow into the reactor coolant system but demonstrates the operation of the valves, pump circuit breakers, and automatic circuitry upon initiation of safety injection.

The accumulator tank pressure and level are continuously monitored during plant operation and flow from the tanks can be checked at any time using test lines.

The accumulators and the safety injection piping up to the final isolation valve are maintained sufficiently filled of borated water at refueling water concentration while the plant is in operation to ensure the system remains operable and performs properly. Flow in each of the high head injection headers and in the main flow line for the residual heat removal pumps is monitored by flow and pressure instrumentation.

The design provided for capability to test initially, to the extent practical, the full operational sequence up to the design conditions for the safety injection system to demonstrate the state of readiness and capability of the system. These functional tests provided information to confirm valve operating times, pump motor starting times, the proper automatic sequencing of load addition to emergency diesel generators, and delivery rates of injection water to the reactor coolant system.

The following general criteria were followed to ensure conservatism in computing the required containment structural load capacity:

1. In calculating the containment pressure, rupture sizes up to and including a double-ended severance of reactor coolant pipe were considered.

2. In considering postaccident pressure effects, various malfunctions of the emergency systems were evaluated. Contingent mechanical or electrical failures were assumed to disable one of the emergency diesel electric generators, two of the fan coolers and one of the containment spray pumps (as discussed in Section 14.3.5.3.7).

3. The pressure and temperature loading obtained by analyzing various loss-of-coolant accidents, when combined with operating loads and maximum wind or seismic forces, does not exceed the load-carrying capacity of the structure, its access opening or penetrations.

Discharge of reactor coolant through a double-ended rupture of the main loop piping, followed by operation of only those engineered safety features, which can run simultaneously with power from two of the three emergency onsite diesel generators, results in a sufficiently low radioactive materials leakage from the containment structure that there is no undue risk to the health and safety of the public.

The concrete containment is not susceptible to a low temperature brittle fracture. The containment liner is enclosed within the containment and thus is not exposed to the temperature extremes of the environs. The containment ambient temperature during operation is between 90°F and 130°F, which is well above the NDTT + 30°F for the liner material. Containment penetrations, which can be exposed to the environment are also designed to the NDTT + 30°F criterion.

The reactor coolant pressure boundary does not extend outside of the containment. Isolation valves for all fluid system lines penetrating the containment provide at least two barriers against leakage of radioactive fluids to the environment in the event of a loss-of-coolant accident. These barriers, in the form of isolation valves or closed systems, are defined on an individual line basis. In addition to satisfying containment isolation criteria, the valving is designed to facilitate normal operation and maintenance of the systems and to ensure reliable operation of other engineered safety features.

After completion of the containment structure and installation of all penetration and weld channels, an initial integrated leakage rate test was conducted at the peak calculated accident pressure, maintained for a minimum of 24 hr, to verify that the leakage rate was not greater than 0.1-percent by weight of the containment volume per day. This leakage rate test was performed using the reference vessel method.

A leak rate test at the peak calculated accident pressure using the same method as the initial leak rate test can be performed at any time during the operational life of the plant, provided the plant is not in operation and precautions are taken to protect instruments and equipment from damage.

Penetrations are designed with double seals so as to permit pressurization of the interior of the penetration whenever a leak test is required. The system utilizes a supply of clean, dry, compressed air which places all the penetrations under an internal pressure above the peak calculated accident pressure (Peak calculated accident pressure is discussed in Section 14.3.5.1.1). Leakage from the system is checked by measurement of the integrated makeup air flow. In the event excessive leakage is discovered, each penetration can then be checked separately.

Capability is provided to the extent practical for testing the functional operability of valves and associated apparatus during periods of reactor shutdown.

Initiation of containment isolation employs coincidence circuits, which allow checking of the operability and calibration of one channel at a time.

The main steam and feedwater barriers and isolation valves in systems, which connect to the reactor coolant system are hydrostatically tested to measure leakage.

Design provisions are made to the extent practical to facilitate access for periodic visual inspection of all important components of the containment air recirculation cooling and containment spray systems.

The containment pressure-reducing systems are designed to the extent practical so that the spray pumps, spray injection valves, spray nozzles can be tested periodically and after any component maintenance for operability and functional performance.

Permanent test lines for the containment spray loop are located so that all components up to the isolation valve at the spray nozzle may be tested. These isolation valves are checked separately.

The air test lines for checking that spray nozzles are not obstructed connect upstream of the isolation valve. Air flow through the nozzles is verified by periodic testing in accordance with the Technical Specifications.

Capability is provided to test initially to the extent practical the operational startup sequence beginning with transfer to alternate power sources and ending with near design conditions for the containment air recirculation cooling and containment spray systems.

Reference sections:

Section Title	Section
Containment System	5
Engineered Safety Features	6
Electrical Systems;	
Design Bases,	8.1
Electrical Systems Design	8.2

1.3.8 Fuel And Waste Storage Systems (GDC 66 - GDC 69)

The new and spent fuel storage racks are designed so that it is impossible to insert assemblies in other than the prescribed locations. The spent fuel storage pit is filled with borated water, normally at a similar concentration to that used in the reactor cavity and refueling canal during refueling operations. The fuel is stored vertically in an array with sufficient neutron absorbers and distance between assemblies to assure k_{eff} <1.0 even if unborated water were used to fill the pit and ≤0.95 when filled with water borated ≥ 2000 ppm boron.

During reactor vessel head removal and while loading and unloading fuel from the reactor, the boron concentration is maintained at not less than that required to maintain the reactor subcritical by 5-percent $\Delta k/k$ with all the rods inserted. The refueling water boron concentration is periodically checked to ensure the proper shutdown margin.

The design of the fuel handling equipment incorporating built-in interlocks and safety features, the use of detailed refueling instructions and observance of minimum operating conditions provide assurance that no incident could occur during the refueling operations that would result in a hazard to public health and safety. The refueling system interlocks are verified to be functioning each refueling shutdown prior to refueling operations.

The refueling water provides a reliable and adequate cooling medium for spent fuel transfer. Heat removal from the refueling water is provided by an auxiliary cooling system.

Adequate shielding for radiation protection is provided during reactor refueling by conducting all spent fuel transfer and storage operations underwater. This permits visual control of the operation at all times while maintaining low radiation levels for periodic occupancy of the area by operating personnel. Pit water level is alarmed and water to be removed from the pit must be pumped out as there are no gravity drains when the pit is isolated from the refueling canal. Shielding is provided for waste handling and storage facilities to permit operation within requirements of 10 CFR 20.

Gamma radiation is continuously monitored in the fuel storage building. A high level signal is alarmed locally and is annunciated in the control room.

Auxiliary shielding for the waste disposal system and its storage components was also designed to limit the dose rate.

All fuel and waste storage facilities are contained and equipment designed so that accidental releases of radioactivity directly to the atmosphere are monitored and will not exceed the applicable limits.

The refueling canal and spent fuel storage pit are reinforced concrete structures with a seam-welded stainless steel plate liner. These structures are designed to withstand any anticipated earthquake loadings as seismic Class I structures so that the liner should prevent leakage even in the event the reinforced concrete develops cracks.

Reference sections:

Section Title	Section
Auxiliary & Emergency Systems;	
Sampling System	9.4
Waste Disposal & Radiation Protection System;	
Waste Disposal System	11.1
Radiation Protection Systems	11.2

1.3.9 Plant Effluents (GDC 70)

Liquid, gaseous, and solid waste disposal facilities are designed so that discharge of effluents and offsite shipments are in accordance with applicable governmental regulations.

Radioactive fluids entering the waste disposal system are collected in sumps and tanks until determination of subsequent treatment can be made. They are sampled and analyzed to determine the quantity of radioactivity, with an isotopic breakdown if necessary. Before any attempt is made to discharge, they are processed as required and then released under controlled conditions. The system design and operation are characteristically directed toward minimizing releases to unrestricted areas. Discharge streams are appropriately monitored and safety features are incorporated to preclude excessive releases.

Radioactive gases are pumped by compressors through a manifold to one of the gas decay tanks where they are held a suitable period of time for decay. Cover gases in the nitrogen blanketing system are reused to minimize gaseous wastes. During normal operation, gases are discharged intermittently at a controlled rate from these tanks through the monitored plant vent. The system is provided with discharge controls so that environmental conditions do not restrict the release of radioactive effluents to the atmosphere. Liquid wastes are processed to remove most of the radioactive materials. The spent resins from the demineralizers and the filter cartridges are packaged and stored onsite until shipment offsite for disposal.

Reference section:

Section Title	Section
Waste Disposal & Radiation Protection System; Waste Disposal System	11.1

REFERENCES FOR SECTION 1.3

1. Letter from P. Zarakas, Con Edison, to H. Denton, NRC, Subject: Actions Taken to Comply with NRC Confirmatory Order of February 11, 1980, dated August 11, 1980.

2. Deleted

1.4 DESIGN PARAMETERS AND PLANT COMPARISON

1.4.1 Design Highlights

The original design of the plant is based upon proven concepts, which have been developed and successfully applied in the construction of pressurized water reactor systems. In subsequent paragraphs, the original design features of the plant are discussed.

1.4.1.1 Power Level

The initial license application power level for Indian Point Unit 2 was 2758 MWt. The increase in this power rating over 1473 MWt for Connecticut Yankee was achieved by a 44-percent increase in heat transfer surface area and a 31-percent increase in average heat flux. The increased heat transfer surface area is due to 22-percent more fuel rods, each 20-percent longer.

The increase in maximum heat flux and the 13.4 kW/ft linear heat generation rate (LHGR) resulting are justified by the results of incore experiments by Westinghouse and others.

1.4.1.2 Reactor Coolant Loops

The reactor coolant system for the Indian Point Unit 2 consists of four loops as compared with three loops for San Onofre and four loops for Connecticut Yankee. The use of four loops in the Indian Point Unit 2 for the production of 2758 MWt requires an attendant increase in the size and capacity of the reactor coolant system components such as the reactor vessel, reactor coolant pumps, piping, and steam generators. These increases represent reasonable engineering extrapolations of existing and proven designs.

1.4.1.3 Peak Specific Power

Based on values for hot channel factors, reactivity coefficients, and other design parameters, which were established in the PSAR and are supported by the previous experience with other plants of the same type, this reactor can be operated safely at power levels at least as high as the license application rating.

1.4.1.4 Fuel Cladding

The fuel rod design for the plant employs zircaloy as a cladding material. This clad has proven successful in numerous operating facilities. The fuel rod dimensions are identical to those in Ginna, Salem, and Zion Station Units 1 and 2.

1.4.1.5 Fuel Assembly Design

The fuel assembly incorporates the rod cluster control concept in a canless 15 x 15 fuel rod assembly using a spring clip grid to provide support for the fuel rods. Extensive out-of-pile and in-pile tests have been performed on this concept; operating experience is available from numerous facilities.

1.4.1.6 Moderator Temperature Coefficient of Reactivity

The reactor has a negative moderator temperature coefficient of reactivity at operating temperature at all times throughout core life (as discussed in Section 14.1.11.2).

1.4.2 IP2 - IP3 Design Differences

An NRC Confirmatory Order (Reference 1) required the Licensees (Consolidated Edison and the Power Authority) to jointly review and identify the significant differences between Indian Point Units 2 and 3 and evaluate these differences in light of the regulatory standards and requirements in existence at the time. Consolidated Edison determined, evaluated, and provided justification for each design difference in submittals to the NRC for acceptance (References 2, 3). These design differences were found acceptable in an NRC Safety Evaluation Report (Reference 4).

REFERENCES FOR SECTION 1.4

1. Letter from Harold R. Denton, NRC, to Consolidated Edison, "Confirmatory Order", dated February 11, 1980.

2. Letter from William J. Cahill, Consolidated Edison, to Harold R. Denton, NRC, "Confirmatory Order", dated May 9, 1980.

3. Letter from John D. O'Toole, Consolidated Edison, to Steven A. Varga, NRC, "Confirmatory Order", dated May 27, 1982.

4. Letter from Steven A. Varga, NRC, to John D. O'Toole, Consolidated Edison, "Confirmatory Order", dated December 1, 1982.

1.5 RESEARCH AND DEVELOPMENT REQUIREMENTS

Research and development were conducted relating to finalization of core design details and parameters, air recirculation system halogen filters, failure of core cooling systems and means to ameliorate consequences, emergency core cooling system, control rod ejection analysis, and reactor coolant pump controlled leakage seals.

The detailed final core design and thermal-hydraulics and physics parameters have been completed. The nuclear design including fuel configuration and enrichments, control rod pattern and worths, reactivity coefficients and boron requirements are presented in Section 3.2.1 and the final thermal-hydraulics design parameters are in Section 3.2.2. Section 3.2.3 presents the final fuel, fuel rod, fuel assembly, and control rod mechanical design. The core design incorporates fixed burnable poison rods[1] in the initial loading to ensure a negative moderator reactivity temperature coefficient at operating temperature. This improves reactor stability and lessens the consequences of a rod ejection or loss-of-coolant accident.

Core stability has been analyzed[2-4] and design provisions for detection and control of potential xenon oscillations have been finalized[3]. The original core design incorporated part-length control rods for controlling these xenon oscillations and shaping the axial power distribution. These have since been found unnecessary and removed from the reactor. X-Y control is not required and therefore not provided. Tests in operating reactors demonstrate the ability of the out-of-core instrumentation to give accurate indication of power redistribution and provide the operator information necessary to monitor redistributions and control axial oscillations by moving the rods in a prescribed pattern[3]. This capability was verified during startup tests in the Indian Point Unit 2 Plant.

Full-size filter tests were conducted for the Connecticut Yankee Atomic Power Company to demonstrate the efficiency for iodine removal under the most extreme conditions anticipated in the postaccident containment environment. The results of these tests[5] filed with the former U. S. Atomic Energy Commission (now the U. S. Nuclear Regulatory Commission) under Docket No. 50-213, are directly applicable to the charcoal filter system originally employed in this plant. The charcoal filters are no longer required by the Technical Specifications and the radiological consequences analysis presented in Section 14.3.6 does not credit the filters.

A program for development of a crucible system design, which would contain the reactor core assuming failure of the core cooling system to prevent a core meltdown, was undertaken. A scheme for containing the molten core in a water submerged high melting point refractory lined steel crucible resulted. Refractory materials and crucible physical design were investigated along with analysis of the temperature distribution expected with the molten core and crucible refractory, and steam and water recirculation paths. As a result of uncertainties in material properties at the high application temperature, the lack of experimental proof that the boiling core mass would dissipate its heat upward through the water cover, and the possibility of violent liquid metal-water reactions, it became apparent that the proper emphasis for research and development on the loss-of-coolant accident should be increased emphasis placed on research and development for emergency core cooling system improvement in order to eliminate need for a crucible. This is supported by the conclusions of the Report of Advisory Task Force on Power Reactor Emergency Cooling, "Emergency Core Cooling," USAEC.

This additional development effort on emergency core cooling system design resulted in the modification of the system to include pressurized accumulator tanks for rapid core reflooding. This increased flooding capability limits the clad temperature after a loss-of-coolant accident to well below the melting temperature of Zircaloy-4, minimizes metal-water reaction and ensures that the core remains in place and intact thereby ensuring preservation of essential heat transfer geometry. The system design incorporates redundancy of components such that the minimum required water addition rates can be met assuming any active component to fail concurrent with the loss-of-coolant accident or, over the long term period of postaccident core decay heat removal a passive or active component failure in either the safety injection or service water systems, or an active failure in the component cooling water system. Details of system design and operation are given in Chapters 6 and 9, and analysis of the loss-of-coolant accident is presented in Section 14.3. Because of the incorporation of this revised emergency core cooling system, the reactor pit crucible was deleted from the plant design. Although it is not required to provide cooling for molten fuel in the bottom of the reactor vessel with the upgraded emergency core cooling system performance, the clearance between the insulation and the instrumentation penetrations with the pressure relief holes in the insulation at the top of the vessel provide assurance that water in the flooded reactor vessel cavity will be in contact with the vessel. No other provisions for direct vessel cooling are provided or required.

A control rod ejection analysis was performed for the final core design, rod worths, rod position limits, and moderator reactivity temperature coefficient. As mentioned above, the addition of the burnable poison rods eliminates power operation with a positive moderator temperature coefficient and reduces the severity of the ejected rod accident, hence, lessening the need for research and development on this subject. The analysis is presented in Section 14.2.6.

The reactor coolant pump controlled leakage seal design for this plant has been fully developed. A full scale mock-up of this seal was operated for over 100 hr to confirm that seal deflection and leak rate under load were acceptable. The full scale mock-up has been used during the development of the controlled leakage seal to provide information related to long-term performance. One of the two seals used in this plant was operated about 300 hr and the other about 100 hr, each in its pump motor unit. During hot functional testing in the plant, before the core was loaded, additional operation brought the total operating time for each seal to well over 500 hr. Successful operation of similar seals had previously been demonstrated with over 5000 hr total running time in San Onofre and over 3000 hr in Haddam Neck.

REFERENCES FOR SECTION 1.5

1. P. M. Wood, E. A. Bassler, et al., Use of Burnable Poison Rods in Westinghouse Pressurized Water Reactors, WCAP-7113, October 1967.

2. P. M. Wood, J. M. Gallagher, R. M. Metz, R. A. Dean, Use of Part-Length Absorber Rods in Westinghouse Pressurized Water Reactors, WCAP-7072, June 1967.

3. Westinghouse Electric Corporation, Power Distribution Control of Westinghouse Pressurized Water Reactors, WCAP-7208, October 1968.

4. Westinghouse Electric Corporation, Power Maldistribution Investigations, WCAP-7407-L (proprietary), January 1970.

5. Connecticut Yankee Charcoal Filter Tests, CYAP-101, December 1966.

1.6 IDENTIFICATION OF CONTRACTORS [Historical Information Only]

The Indian Point Unit 2 was designed and built by the Westinghouse Electric Corporation as prime contractor for Con Edison. Westinghouse undertook to provide a complete, safe, and operable nuclear power plant ready for commercial service. The project was directed by Westinghouse from the offices of its Atomic Power Division in Pittsburgh, Pennsylvania, and by Westinghouse representatives at the plant site during construction and plant startup. Westinghouse engaged United Engineers and Constructors of Philadelphia, Pennsylvania, to provide the design of certain portions of the plant.

The plant construction was under the general direction of Westinghouse through United Engineers and Constructors, which was responsible for the management of all site construction activities and either performed or subcontracted the work of construction and equipment erection. Preoperational testing of equipment and systems at the site and initial plant operation was performed by Con Edison personnel under the technical direction of Westinghouse.

1.7 PROJECT REORGANIZATION - DECEMBER 1969 [Historical Information Only]

This section describes a reorganization in project management, which was implemented by Westinghouse in December 1969.

Westinghouse formed a wholly-owned subsidiary corporation, called WEDCO Corporation, to perform certain functions at the Indian Point site of Con Edison. Westinghouse remained the prime contractor and continued to exercise overall control and to have full responsibility for the Indian Point 2 project. WEDCO performed, under Westinghouse, project management, engineering, quality assurance, construction, and procurement functions for Indian Point Unit 2. These functions were previously carried out by Westinghouse or United Engineers and Constructors (UE&C).

The entire Westinghouse senior management organization, which prior to the advent of WEDCO, was responsible for the Westinghouse effort at Indian Point Unit 2, remained responsible. All other personnel within Westinghouse senior management who, prior to WEDCO, carried any responsibility in any area for Indian Point Unit 2 continued to carry those responsibilities, regardless of the formation of WEDCO or changes in title or designation. Furthermore, WEDCO had behind it the full organization and strength of the Westinghouse

Electric Corporation: Westinghouse engineering, legal, and other personnel continued to be available for the project.

The functional relationships among Con Edison, Westinghouse, WEDCO, and UE&C are shown in Historical Figure 1.7-1.

Westinghouse WEDCO-UE&C Relationship.

Westinghouse retained UE&C as its architect-engineer-constructor to perform certain work and services in connection with the plant. Initially, UE&C performed services within its scope in the following areas:

1. Design and Engineering

2. Procurement

3. Construction Management and Construction

4. Quality Assurance (including Home Office Quality Control Engineering, Vendor Surveillance and Onsite Quality Control)

Westinghouse removed items (2) and (3) from the scope of work to be performed by UE&C and assigned these functions to WEDCO. In these areas, however, UE&C provided qualified personnel to assist in effectuating the transition of work to Westinghouse and WEDCO.

UE&C continued to have responsibility for all of the design and engineering functions and all of the quality assurance functions, including home office quality control engineering, vendor surveillance and onsite quality control, for which it had responsibility prior to the advent of WEDCO. UE&C continued to have direct corporate responsibility to Westinghouse for all of the work within its scope.

In its organizational structure, WEDCO exercised a high level quality and engineering reliability function. This function included the activities previously performed by the Nuclear Power Service Staff Resident Quality Assurance Engineer, and in addition included the centralization and overall management for quality assurance activities previously performed by various organizations. This function was carried out by a Reliability Manager who was located at the site. The Reliability Manager was responsible for surveillance visits to selected shops or suppliers. This function was previously delegated to the Westinghouse Nuclear Power Services Group. In addition, the Reliability Manager continually audited the quality assurance efforts of UE&C. In effect, a new reliability management function over and above those previously set forth was established while all existing organizational functions and responsibilities for quality assurance were maintained.

The quality control functions previously performed at various Westinghouse organizational levels continued to be performed. At the Westinghouse headquarters level, the staff quality assurance audit team reviewed periodically the quality control program for Indian Point Unit 2 as it had done in the past. At the Westinghouse PWR Systems Division level, the quality control functions performed by that division for the nuclear steam supply system continued as before.

Con Edison.

The project reorganization described did not in any way alter the ultimate responsibility of Con Edison for the quality assurance program. There was no basic change in the Con Edison program. However, the following minor procedural changes were made in view of the existence of WEDCO:

1. Con Edison's monitoring function included monitoring the activities of WEDCO.

2. Con Edison forwarded the United States Testing Company quality assurance reports to Westinghouse and/or WEDCO.

3. Con Edison contacted Westinghouse and/or WEDCO for necessary corrective action.

1.7 FIGURES

Figure No.	**Title**
Figure 1.7-1	Functional Relationships [Historical]

1.8 PROJECT REORGANIZATION - MARCH 1970 [Historical Information Only]

This section describes a change, which was implemented by Westinghouse in the spring of 1970 in the project organization.

The changes made in December 1969 (see Section 1.7) involved the creation of WEDCO and the delegation to WEDCO by Westinghouse of certain functions at the Indian Point site previously carried out by Westinghouse or United Engineers and Constructors (UE&C). Following the December 1969 reorganization, UE&C retained the following functions within its scope of work as architect-engineer-constructor:

1. Design and Engineering.

2. Quality Assurance (including Home Office Quality Control Engineering, Vendor Surveillance, and Onsite Quality Control).

The change consisted of the removal of the vendor surveillance and onsite quality control portions of item (2) from the scope of work to be performed by UE&C, and the assigning of these functions to WEDCO.

There was little change of personnel involved in the transfer of the onsite quality control function. WEDCO employed a Manager of Vendor Surveillance and other personnel for this work. The transition in this respect was gradual. New personnel were phased in and the UE&C personnel were used during the transition period to ensure continuity of the surveillance program. The transfer was made on a purchase-order-by-purchase-order basis, with UE&C personnel working with new personnel in performing the surveillance during the transition.

To assure that a level of quality assurance review was not lost, the organization of the WEDCO reliability group was structured to provide for an independent, internal audit of the two quality assurance functions transferred to WEDCO. The Vendor Surveillance Group and Onsite Quality Control Group each reported directly to the Reliability Manager.

The activities of both the Vendor Surveillance Group and the Onsite Quality Control Group were audited by a Systems Reliability Group. The Systems Reliability Group reported directly to the Reliability Manager to ensure its functional independence. Historical Figure 1.8-1 shows these organizational relationships in chart form.

1.8 FIGURES

Figure No.	**Title**
Figure 1.8-1	Organization Chart WEDCO Reliability Group [Historical]

1.9 SUPPLEMENTS AND REVISIONS TO ORIGINAL FSAR

1.9.1 Supplements

Supplement 1 to the Indian Point Unit 2 Final Safety Analysis Report consisted of responses to questions from the Atomic Energy Commission as contained in two letters. The first letter from Peter A. Morris, Director of the Division of Reactor Licensing, on March 5, 1969, to Mr. Donham Crawford of Con Edison, requested additional information on the medical plans and facilities at Indian Point. The questions and responses are found following Tab I of Volume 5 of the original FSAR. These responses were incorporated into Section 11.2.5 of the original FSAR as page changes. The responses to the questions in Volume 5 indicate where the specific answer may be found in the page change.

The second letter to Arthur N. Anderson of Con Edison from Peter A. Morris, dated August 4, 1969, requested additional information on Chapters 1, 2, 3, 4, 5, 6, 7, 8, 11, 12, and 14 of the original FSAR. Supplement 1 responded to several of the questions in the second letter found behind Tab II of Volume 5 of the original FSAR. The responses consisted of questions and answers given in Volume 5 of the original FSAR and also of page changes to the original text of the FSAR in some instances.

Supplement 2 to the Indian Point Unit 2 Final Safety Analysis Report consisted of responses to questions from the Atomic Energy Commission and page changes to the report. The questions were contained in a letter to Arthur N. Anderson of Con Edison from Peter A. Morris, Director of the Division of Reactor Licensing, dated August 4, 1969. The responses consisted of questions and answers added to Volume 5 of the original FSAR in the proper order behind Tab II. Page changes for the FSAR were included with Supplement No. 2.

Supplement 3 to the Indian Point Unit 2 Final Safety Analysis Report consisted of responses to questions from the Atomic Energy Commission and page changes to the report. The questions were contained in a letter to Arthur N. Anderson of Con Edison from Peter A. Morris, Director of the Division of Reactor Licensing, dated August 4, 1969. The responses consisted of questions and answers added to Volume 5 of the original FSAR in the proper order behind Tab II. This supplement responded to several questions concerning Chapters 1, 4, 5, 7, 8, and 11 of the report.

Supplement 4 to the Indian Point Unit 2 Final Safety Analysis Report consisted of responses to questions from the Atomic Energy Commission and page changes to the report. The questions were contained in a letter to Arthur N. Anderson of Con Edison from Peter A. Morris, Director of the Division of Reactor Licensing, dated August 4, 1969. The responses consisted of questions and answers added to Volume 5 of the original FSAR in the proper order behind Tab II. Also included with this supplement was a description of the project reorganization within Westinghouse. This supplement also responded to several questions concerning Chapters 4, 5, 7, 11, and 14 of the report.

Supplement 5 to the Indian Point Unit 2 Final Safety Analysis Report consisted of responses to questions from the Atomic Energy Commission. The questions were contained in a letter to Arthur N. Anderson of Con Edison from Peter A. Morris, Director of the Division of Reactor Licensing, dated August 4, 1969, and a letter to William J. Cahill, Jr., of Con Edison from Peter A. Morris dated November 13, 1969. The responses consisted of questions and answers added to Volume 5 of the original FSAR in the proper order behind Tab II. The supplement responded to several questions concerning Chapters 1, 4, 6, 11, 12, and 14 of the report.

Supplement 6 to the Indian Point Unit 2 Final Safety Analysis Report consisted of responses to questions from the Atomic Energy Commission. The questions were contained in a letter to Arthur N. Anderson of Con Edison from Peter A. Morris, Director of the Division of Reactor Licensing, dated August 4, 1969, and a letter to William J. Cahill, Jr., of Con Edison from Peter A. Morris dated November 13, 1969. The responses consisted of questions and answers added to Volume 5 of the original FSAR in the proper order behind Tab II. The supplement responded to several questions concerning Chapters 1, 3, 4, 6, 9, and 14 of the report. Also included with this supplement was the Indian Point Unit 2 Containment Design Report.

Supplement 7 to the Indian Point Unit 2 Final Safety Analysis Report consisted of responses to questions from the Atomic Energy Commission and page changes to the report. The questions were contained in a letter to Arthur N. Anderson of Con Edison, from Peter A. Morris, Director of the Division of Reactor Licensing, dated August 4, 1969, and a letter to William J. Cahill, Jr., of Con Edison, from Peter A. Morris, dated November 13, 1969. This supplement responded to several questions concerning Chapters 4, 5, 6, 9, 13, and 14 of the report.

Supplement 8 to the Indian Point Unit 2 Final Safety Analysis Report consisted of responses to questions from the Atomic Energy Commission and page changes to the report. The questions were contained in a letter to Arthur N. Anderson of Con Edison, from Peter A. Morris, Director of the Division of Reactor Licensing, dated August 4, 1969, and a letter to William J. Cahill, Jr., of Con Edison, from Peter A. Morris, dated November 13, 1969. The responses consisted of questions and answers added to Volume 5 of the original FSAR. The supplement responded to questions concerning Chapters 4, 6, 7, and 13 of the report.

Supplement 9, 10, 12, 14, 20 and 21 to the Indian Point Unit 2 Final Safety Analysis Report consisted of corrections and additional information for the original FSAR in the form of page changes.

Supplement No. 11 to the Indian Point Unit 2 Final Safety Analysis Report provided the proposed Technical Specifications for operation of the facility in accordance with the rules of practice, 10 CFR 50.36.

Supplement 13 to the Indian Point Unit 2 Final Safety Analysis Report consisted of responses to questions from the Atomic Energy Commission contained in a letter from Peter A. Morris, Director of the Division of Reactor Licensing, on July 24, 1970, to William J. Cahill, Jr., of Con Edison. The letter requested additional information on Chapters 1, 4, 7, 8, 12, and 14 of the original FSAR. The responses consisted of questions and answers given in Volume 5 of the FSAR and also of page changes to the original text of the FSAR in some instances.

Supplement 15 to the original Final Safety Analysis Report consisted of correction pages that updated certain areas where final design parameters were available and where design modifications had resulted from AEC review. In addition, a cross-reference index was submitted for each chapter of the FSAR where required. The index referenced the responses to questions in Volumes 5 and 6 where additional information could be found concerning specific sections. The proposed Technical Specifications were reissued in their entirety with this supplement. This issue superseded the specifications submitted in Supplement 11.

Supplement 18 to the original Final Safety Analysis Report consisted of the relocation of information from the site Custom Technical Specifications into the UFSAR for items and topics that were no longer found in the Improved Technical Specifications. It also updated references to the new Technical Specification sections, to information relocated from the Technical Specifications into the Off Site Dose Calculation Manual (ODCM) and added cross references to the new Technical Requirements Manual (TRM).

Supplement 19 to the original Final Safety Analysis Report consisted of corrections and additional information for the original FSAR in the form of changes to reflect several plant modifications, changes to reflect 10 CFR 100.11, the new fuel design and new core design for Cycle 17 and Cycle 16 Core Reload Design, the permanent increase in Tave to 565^{o}F, and the approved alternate source term fuel handling accidents (FHB & VC) which take no credit for charcoal filtration. Changes were also included from NRC approved projects, including Appendix “K” Power Uprate [1.4% Power Uprate] with the re-analysis of some of the Chapter 14 accidents to account for the 1.4% power uprate, re-analysis of the Loss of Electrical Load transients and LONE/LOOP transients, and the re-analysis of the Feedwater System Malfunction with a step increase of 120% of nominal feedwater flow to one steam generator, and to reflect the approved Stretch Power Uprate to 3216 MWt.

1.9.2 Revisions

Pursuant to 10 CFR 50.71(e), Con Edison submitted an updated Final Safety Analysis Report for Indian Point Unit 2 on July 22, 1982, reflecting changes made up to a maximum of 6 months prior to the submittal date. In addition, the following revisions to the updated Final Safety Analysis Report have been submitted to date:

Revision 1, July 1983
Revision 2, July 1984
Revision 3, July 1985
Revision 4 July 1986
Revision 5, June 1987
Revision 6, June 1988
Revision 7, June 1989
Revision 8, June 1990
Revision 9, June 1991

Revision 10, June 1992
Revision 11, June 1993
Revision 12, June 1994
Revision 13, December 1995
Revision 14, December 1997
Revision 15, December 1999
Revision 16, July 2001
Revision 1 May 2003
Revision 18, October 2003
Revision 19, May 2005
Revision 20, November 2006
Revision 21, October 2008
Revision 22, October 2010
Revision 23, October 2012

Revision 2, in addition to reflecting required changes, incorporated a major editorial effort to standardize the format of the updated FSAR and to correct typographical, grammatical, and syntax errors, so that the material is presented in a more uniform and clear manner. The changes of technical content and some major editorial changes were marked in the margins with the numeral 2. The majority of editorial changes were minor and were not marked individually. A changed page, however, was indicated by the label Revision 2 at the lower right hand corner.

1.10 QUALITY ASSURANCE PROGRAM

1.10.1 General

Entergy's Quality Assurance Program (QAP) for Indian Point Unit 2 is in accordance with the quality assurance requirements of 10 CFR 50 Appendix B. The QAP is described in a Quality Assurance Program Manual, which satisfies the criteria of Appendix B. Changes to the program description are submitted to the NRC in accordance with the provisions of 10 CFR 50.54(a)(3).

1.10.2 Scope

The Quality Assurance Program provides control for activities affecting the quality of structures, systems, and components of the plant and their operation to the extent consistent with their importance to safety. Those structures, systems, and components of the plant that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public are designated "Class A" as described in the Quality Assurance Program. All items and activities affecting safety addressed in Regulatory Guide 1.29 “Seismic Design Classification” revision 3, September 1978, are also governed by the Quality Assurance Program. A list of Class A items is maintained. Elements of the Quality Assurance Program are also applicable to activities and items affecting safety as defined in Licensing commitments.[2]

It is recognized that not every portion of each of the listed systems and components affect the safety related function. Therefore, allowance is made for subcomponents of systems to be declassified. When such is the case, the agreement is appropriately documented identifying the parts or subcomponents concerned and showing appropriate concurrences.

1.10.3 Organization And Responsibilities

The major organizations or groups participating in the Quality Assurance Program are: Nuclear Quality Assurance and Oversight, Nuclear Power, Nuclear Power Engineering, and the Safety Review Committee. The duties and responsibilities of the individuals participating in the Quality Assurance Program are described in procedures, or manuals.

REFERENCES FOR SECTION 1.10

1. Deleted
2. Letter from John D. O'Toole, Con Edison, to Director of Nuclear Reactor Regulation, NRC, Subject: Response to NRC letter of September 23, 1980 to Mr. Zarakas requesting information on the Quality Assurance Program for Indian Point Unit 2 dated March 11, 1981.

TABLE 1.10-1
DELETED

1.11 DESIGN CRITERIA FOR STRUCTURES AND COMPONENTS

1.11.1 Definition Of Seismic Design Classifications

All structures and components are classified as seismic Class I, Class II, or Class III as recommended in:

1. TID-7024, "Nuclear Reactors and Earthquakes," August 1963 and,

2. G. W. Housner, "Design of Nuclear Power Reactors Against Earth-quakes," Proceedings of the Second World Conference on Earthquake Engineering, Volume I, Japan, 1960, Pages 133, 134 and 137.

Class I

Seismic Class I is defined as those structures and components including instruments and controls whose failure might cause or increase the severity of a loss-of-coolant accident or result in an uncontrolled release of radioactivity causing more than 10 rem to the thyroid or 10 rem whole body to the average adult beyond the nearest site boundary. Also included are those structures and components vital to safe shutdown and isolation of the reactor.

Class II

Class II is defined as those structures and components, which are important to reactor operation but not essential to safe shutdown and isolation of the reactor and whose failure could result in the release of radioactivity causing more than 1.0 rem to the thyroid or 0.5 rem whole body dose to the average adult beyond the nearest site boundary.

Class III

Class III is defined as those structures and components, which are not directly related to reactor operation or containment. In Indian Point Unit 2, the only portions of the plant, which are not seismic Class I and which might carry substantial radioactivity because of required safeguards operation or requirements for safe shutdown and isolation of the reactor are portions of the chemical and volume control system and waste disposal system.

The specific components in the chemical and volume control system are the volume control tank, holdup tank, and the concentrates holding tank with associated piping, valves and supports. These components are all seismic Class I. In addition, the design of the system tanks and their location were based upon the commitment that a vessel rupture would not cause doses in excess of 10 CFR 20 limits at the exclusion radius.

The specific components in the waste disposal system are the gas decay tanks with their associated piping, valves and supports. These components are all seismic Class I. In addition, the gas decay tanks of the waste disposal system have been designed such that the failure of any tank will not exceed 10 CFR 20 doses at the exclusion radius.

The analysis showing that the rupture of the volume control tank or a gas decay tank does not exceed the special dose limits selected for Indian Point Unit 2 is found in Section 14.2.3 of the FSAR.

Those components of the chemical and volume control system that are not seismic Class I are as follows: batching tank, monitor tanks, monitor tank pumps, chemical mixing tank, and resin fill tank. In addition, the boric acid evaporator and the condensate demineralizer are not seismic Class I.

Those components of the waste disposal system, which are not seismic Class I include: waste condensate tank and pumps.

Failure of these components will not result in offsite doses in excess of 10 CFR 20 limits at the site exclusion radius.

All components, systems, and structures classified as seismic Class I are designed in accordance with the following criteria:

1. Primary steady state stresses, when combined with the seismic stress resulting from the response to a ground acceleration of 0.05g acting in the vertical and 0.10g acting in the horizontal planes simultaneously, are maintained within the allowable stress limits accepted as good practice and, where applicable set forth in the appropriate design standards, e.g., ASME Boiler and Pressure Vessel Code, USAS B31.1 Code for Pressure Piping, ACI 318 Building Code Requirements for Reinforced Concrete, and AISC Specifications for the Design and Erection of Structural Steel for Buildings.

2. Primary steady state stresses when combined with the seismic stress resulting from the response to a ground acceleration of 0.10g acting in the vertical and 0.15g acting in the horizontal planes simultaneously, are limited so that the

function of the component, system or structure shall not be impaired as to prevent a safe and orderly shutdown of the plant.

All Class II structures and components are designed on the basis of a static analysis for a ground acceleration of 0.05g acting in the vertical and 0.10g acting in the horizontal directions simultaneously.

The structural design of all Class III structures meets the requirements of the applicable building code, which is the "State Building Construction Code" State of New York, 1961. This code does not reference the Uniform Building Code.

The Original Steam Generator Storage Facility (OSGSF) has been constructed for the storage of the original steam generators. The OSGSF is a seismic Class III structure, designed in accordance with the requirements of the State of New York Official Compilation of Codes, Rules and Regulations, Title 9, Subtitle S, 1995 edition, copyright 1999, and the American Concrete Institute (ACI) 318, Building Code Requirements for Structural Concrete, 1999.

Table 1.11-1 gives the damping factors used in the design of components and structures.

The design of seismic Class I structures and components utilizes the "response spectrum" approach in the analysis of the dynamic loads imparted by the earthquake. The analysis is based upon the response spectra shown on Figures 1.11-1 and 1.11-2.

The following method of analysis is applied to seismic Class I structures and components, including instrumentation:

1. The natural period of vibration of the structure or component is determined.

2. The response acceleration of the component to the seismic motion is taken from the response spectrum curve at the appropriate period.

3. Stresses and deflections resulting from the combined influence of normal loads and the seismic load due to the design earthquake (0.05g acting in the vertical and 0.10g acting in the horizontal planes simultaneously) are calculated and checked against the limits imposed by the design standard.

4. Stresses and deflections resulting from the combined influence of normal loads and the seismic loads due to the maximum potential earthquake (0.10g acting in the vertical and 0.15g acting in the horizontal planes simultaneously) are calculated and checked to verify that deflections do not cause loss of function and that stresses do not produce rupture.

   Where the vibrator system is of a highly complex geometric shape, such as piping systems, the maximum response from the response curve with the appropriate damping factor is selected. By using this conservative value and demonstrating that the stresses are satisfactory, it becomes unnecessary to perform any further analysis to determine the natural periods of the system.

For a further discussion of the models and methods used for the seismic Class I design of structures, equipment, piping, instrumentation and controls, see Section 1.11.4.

1.11.2 Classification Of Particular Structures And Equipment

Examples of particular structure and equipment classifications are given below. These classifications are not intended to be all-inclusive.

Item	Class
Buildings and Structures	
Containment (including all penetrations and airlocks, the concrete shield, the liner, and the interior structures)	I
Spent fuel pit	I
Control Building	I
Diesel Generator Building	I
Intake structure (to the extent that water is always available to the service water pumps)	I
Service water screenwell	I
Primary Auxiliary Building	I
Turbine Building	III
Buildings containing conventional facilities Such as the Maintenance and Outage Building	III
Original Steam Generator Storage Facility	III

Equipment, Piping, and Supports

[***Note*** *- Class I components (equipment, piping, instrumentation, etc.) located in or supported on a Class II structure are protected from earthquake damage or are backed up by other Class I components located in or supported by a Class I structure.*]

Item	Class
Reactor control and protection system	I
Radiation monitoring system	I
Process instrumentation and controls	I
Reactor Vessel and its supports	I
Vessel internals Fuel assemblies Rod cluster control assemblies and drive mechanisms Supporting and positioning members	I

Incore instrumentation structure	
Reactor coolant system	I
Piping and valves (including safety and relief valves)	
Steam generators	
Pressurizer	
Reactor coolant pumps	
Supporting and positioning members	
Main Steam system, up to and including the isolation valve	I
Engineered safety features	I
Safety injection system (including safety injection and residual heat removal pumps, refueling water storage tank, accumulator tanks, residual heat removal heat exchangers and connecting piping and valving)	
Containment spray system (including spray pumps, spray headers, and connecting piping and valving)	
Containment air recirculation cooling system (including fans, coolers, ducts, valves, and demisters)	
Auxiliary building ventilation system	I
Condensate storage tanks	I
Pressurizer relief tank	II
Residual heat removal loop	I
Containment penetration and weld channel pressurization system	I
Component cooling loop	I
Instrument air system (essential sections)	I
Isolation valve seal water system	I
Sampling system	II
Spent fuel pit cooling loop	II
Fuel transfer tube	I
Emergency power supply system	I
Diesel generators and fuel oil storage tank	I
DC power supply system	
Power distribution lines to equipment required	

for transformers and switchgear supplying the engineered safety features	
Control panel boards	
Motor control centers	
Control Equipment, facilities and lines necessary for the above seismic Class I items	I
Waste disposal system	I
Chemical drain tank	
Waste holdup tanks	
Gas decay tanks	
Reactor coolant drain tank	
Compressors	
Waste holdup tank pumps	
Interconnecting waste gas piping	
Waste disposal system	II or III
All elements not listed as seismic Class I	
Containment crane	I
Manipulator and other cranes	III
Conventional equipment, tanks and piping, other than Classes I and II	III
Auxiliary boiler feed and service water pumps and piping	I
The chemical and volume control system is considered seismic Class I except for the components listed below:	
Batching tank	II
Monitor tanks	III
Monitor tank pumps	III
Chemical mixing tank	II
Resin fill tank	III

1.11.3 Design Criteria For Seismic Class I Structures And Equipment

The criteria for functional adequacy of structures, equipment, piping, instrumentation, and controls follow.

No loss of function implies that rotating equipment will not freeze, pressure vessels will not rupture, supports will not collapse under the load, systems required to be leaktight will remain leaktight, and components required to respond actively (such as valves and relays) will respond actively.

The criteria for functional adequacy of the structures state stresses will not exceed yield when subjected to a 0.15g ground acceleration. The manner in which these criteria have been met is by limiting stresses in seismic Class I structures to meet the above criteria.

For all seismic Class I piping and their supports, the criteria for functional adequacy and the manner in which the criteria are met are the following:

with a ground acceleration of 0.15g horizontal, the spectral acceleration corresponding to the maximum point on the 0.5-percent critical damping response curve was used to calculate an equivalent static force imparted to the pipe at its support points. This resulted in a seismic design load approximately equal to 0.6W horizontally and 0.4W vertically taken simultaneously, where W is the weight of the pipe including static forces. The sum of the resulting additional stress plus the normal stresses was limited to 1.2 times the B31.1 code allowable. The stresses in the pipe supports and hangers were likewise limited to 1.2 times the B31.1 code allowable.

Since all the buildings containing seismic Class I piping are essentially rigid structures, no amplification is expected.

For seismic Class I equipment and tanks the same method was used to arrive at an equivalent static force. In each case, the total of seismic and normal stresses was limited to the applicable code allowable. The refueling water storage tank and condensate storage tank were designed in accordance with the stress limitations of American Water Works Association Standard D100.All components of the reactor coolant system and associated systems are designed to the standards of the applicable ASME code or USAS code. The loading combinations, which are employed in the design of seismic Class I components of these systems, i.e., vessels, piping, supports, vessel internals and other applicable components, are given in Table 1.11-2.

Table 1.11-2 also indicates the stress limits, which are used in the design of the listed equipment for the various loading combinations. The original design criteria given above and in Table 1.11-2 have been modified in certain instances in accordance with NRC guidance given in References 3 and 4. Generic Letter 87-11 allows for the elimination of pipe whip restraints and jet impingement shields, which were installed to mitigate the effects of arbitrary intermediate pipe ruptures, provided certain criteria are met.

These design criteria have also been modified in certain instances by the application of "leak before break" technology, as discussed in Section 4.1.2.4.

To be able to perform their function, i.e., allow core shutdown and cooling the reactor vessel, internals must satisfy deformation limits, which are more restrictive than the stress limits shown in Table 1.11-2. For this reason the reactor vessel internals are treated separately.

1.11.3.1 Piping, Vessels, and Supports

The reasoning for selection of the load combinations and stress limits given in Table 1.11-2 is as follows. For the design earthquake, the nuclear steam supply system is designed to be capable of continued safe operation, i.e., for the combination of normal loads and design earthquake loading. Critical equipment and supports needed for this purpose are required to operate within normal design limits as shown in line 2 of Table 1.11-2.

In the case of the maximum potential earthquake, it is only necessary to ensure that critical components do not lose their capability to perform their safety function, i.e., shut the plant down and maintain it in a safe condition. This capability is ensured by maintaining the stress limits as shown in line 3 of Table 1.11-2. No rupture of a seismic Class I pipe can be caused by the occurrence of the maximum potential earthquake. With respect to the seismic design of the piping supports, relative displacement between anchor points has been considered in the seismic analysis of the main steam lines, where largest relative displacements are expected. Analysis indicates that the stress at the highest seismically stressed point is affected by less than 10-percent when relative anchor displacements are considered. The seismic supports installed have been verified to agree with the design location and therefore the locations used in the analyses.

Careful design and thorough quality control during manufacture and construction and periodic inspection during plant life, ensures that the independent occurrence of a reactor coolant pipe rupture is extremely remote. If it is assumed that a reactor coolant pipe ruptures, the stresses in the unbroken leg will be as noted in line 4 of Table 1.11-2.

1.11.3.2 Reactor Vessel Internals

1.11.3.2.1 Design Criteria for Normal Operation

The internals and core are designed for normal operating conditions and subjected to loads of mechanical, hydraulic, and thermal origin. The response of the structure under the design earthquake is included in this category.

The stress criteria established in Section III of the ASME Boiler and Pressure Vessel Code, Article 4, have been adopted as a guide for the design of the internals and core with exception of those fabrication techniques and materials, which are not covered by the code, such as the fuel rod cladding. Seismic stresses are combined in the most conservative way and are considered primary stresses.

The members are designed under the basic principles of: (1) maintaining distortions within acceptable limits, (2) keeping the stress levels within acceptable limits, and (3) prevention of fatigue failures.

1.11.3.2.2 Design Criteria for Abnormal Operation

The abnormal design condition assumes blowdown effects due to a reactor coolant pipe double-ended break.

For this condition the criteria for acceptability are that the reactor be capable of safe shutdown and that the engineered safety features are able to operate as designed. Consequently, the limitations established on the internals for these types of loads are concerned principally with the maximum allowable deflections. The deflection limits for critical internals structures under abnormal operation are presented in Table 14.3-14.

1.11.3.3 Reactor Vessel

The criteria for movement of the reactor vessel, under the worst combination of loads, i.e., normal plus the maximum potential earthquake or normal plus reactor coolant pipe rupture loads, ensure that the radial movement of the reactor vessel will not exceed the clearance between the reactor coolant piping and the surrounding concrete.

The relative motions between reactor coolant system components are controlled by the structures, which are used to support the reactor vessel, the steam generators, the pressurizer and the reactor coolant pumps.

The maximum movement of the reactor vessel under the worst combination of loads, i.e., normal plus maximum potential earthquake or normal plus reactor coolant pipe rupture loads comprises an end deflection on the safety injection piping, which is small even in comparison with that resulting from thermal growth during plant heatup, and is well within the flexibility of the design of the piping system.

The supports are designed to limit the stresses in the pipe to the stress limits given in Table 1.11-2.

1.11.4 Models And Methods For Seismic Class I Design

The variety of design problems associated with the seismic analysis of all Class I structures, systems and equipment were approached by various methods. For the design of the reactor, recirculating pumps, and Class I piping an amplification factor of 4.0 was used with respect to ground motion of 0.15g. This amplification factor was based on the maximum for a one-half percent damping of the ground response spectrum. The fundamental frequency of the reactor building internal structure is approximately 17 cycles/sec. As can be seen from Figure 1.11-2 for this frequency level, no significant building amplification of the ground response is encountered.

With the exception of the containment, primary auxiliary building, and electrical cable tunnel, no dynamic analyses were performed on Indian Point Unit 2 structures, hence no mathematical models were developed. The following methods were used in the seismic design of Class I structures.

1.11.4.1 Containment Building

See Sections 2.0, 3.0, and 4.0 of the Containment Design Report for Indian Point Unit 2 containment building structures and components.

1.11.4.1.1 Steel

In the design of the steel, 100-percent of the dead load and 50-percent of the live load were considered. The peak of the response curve for 0.15g ground acceleration and 1.0-percent critical damping was used to obtain the seismic forces, which were distributed by the method described in the Containment Design Report and resisted by the bracing. The 1.0-percent critical damping is conservative since the structure is shop welded and field bolted to the columns. The actual critical damping value would be between 1.0-percent (welded) and 2.5-

percent (bolted). A one-third increase over working stress was allowed in the design of the bracing.

1.11.4.1.2 Concrete

In the design of the concrete, 100-percent of the dead load and 50-percent of the live load were considered. The Modified Rayleigh Method was used to calculate the natural period and the base shear was distributed by the same method described in the Containment Design Report. The forces determined from the response curve for a 0.15g ground acceleration with 5-percent critical damping were applied at the node points where the masses were lumped for the Rayleigh approach. These loads were resisted by the vertical walls, which acted as shear walls, and horizontal reinforcing, which resisted the moment. The Ultimate Strength Design method of ACI 318-63 was used for the design and construction of the containment building.

1.11.4.2 Control Building

The dead load and equipment loads were considered. The period was determined from the formula T = 0.1 n, where n = number of stories (Design of Multistory Reinforced Concrete Building for Earthquake Motions by N. M. Newmark, et. al.). The response curve for 0.15g ground acceleration with 2.5-percent critical damping was used to determine the base shear. This base shear was distributed at the floor levels by the same method described in the Containment Design Report and resisted by a rigid frame structure with a one-third increase on allowable working stresses. The design was controlled by a deflection limitation due to the adjacent Unit 1 control building.

1.11.4.3 Diesel Generator Building

Due to the light weight of the structure, the wind load controlled the design.

1.11.4.4 Fan House

One hundred percent of the dead load and 50-percent of the live load were considered. The peak of the response curve for 0.15g ground acceleration with 5-percent critical damping was used for the concrete structure and the corresponding 2.5-percent was used for the steel superstructure. A one-third increase in allowable working stresses was allowed.

1.11.4.5 Boric Acid Evaporator Building

One hundred percent of the dead load was considered. For method of design, see fan house. Without allowing a one-third stress increase for seismic design, the controlling factor for reinforcing design was the minimum temperature steel requirements of the ACI-318 Building Code.

1.11.4.6 Intake Structure

One hundred percent of the live and dead load were considered. The peak of the response curve for 0.1g (OBE) ground acceleration with 5-percent critical damping was used to obtain the seismic loads. The effect of water sloshing was considered in the earthquake analysis (per TID-7024 "Nuclear Reactors and Earthquakes," Section 6.5). Although DBE was not explicitly considered in the calculation (the seismic forces used in the design shows that DBE is not governing), the controlling factor in the design of the intake structure was the service load with the worst combination being one chamber empty and the adjacent chamber filled with water.

1.11.4.7 Waste Holdup Tank Pit

One hundred percent of the dead load and 50-percent of the live load were considered (including the tank dead weight on the roof). The peak of the response curve for 0.15g ground acceleration with 5-percent damping was used to determine the base shear. Using working stress limits for the seismic design, service loads controlled the design of the top slab. The bottom slab and wall of the pit were designed for earthquake loads with stresses limited to yield multiplied by the Φ factors recommended in Section IV-B of the ACI-318-63 "Building Code." Consideration was given to the tanks in the pit when designing the base slab.

1.11.4.8 Spent Fuel Pit

The seismic loads, as determined in TID-7024 "Nuclear Reactors and Earthquakes," Section 6.5, were resisted by the reinforced concrete walls and base slab. Working stresses were used except for the moment at the base of the walls where ultimate strength design was considered with stresses limited to $\varnothing$fy. The effects of water in the pool are accounted for in this design approach. Ground acceleration of 0.15g was used. In 1990, new high density spent fuel storage racks were installed. Prior to their installation, the spent fuel pit was reanalyzed (Ref. 6). The new racks were also analyzed (Ref. 5 and 6).

1.11.4.9 Electrical Penetration Tunnel

The peak of the response curve for 0.15g ground acceleration with 5-percent critical damping was considered using working stress design limits. The load was considered to act at 2/3 L, where L = the height of the tunnel. Temperature of steel considerations controlled the design of the concrete while service loads controlled the structural steel.

1.11.4.10 Pipe Penetration Tunnel

One hundred percent of the dead load, plus 50-percent of the live load were considered. The peak of the response curve for 0.15g ground acceleration with 5-percent damping was used to find the shear, which was considered as a concentrated load applied at the top slab of the tunnel. A one-third increase on working stress allowables was used in the design.

1.11.4.11 Electrical Cable Tunnel

One hundred percent of the dead load, 50-percent of the surcharge, and 50-percent of live load in the tunnel were considered. The Modified Rayleigh Method was used to determine the natural period and the loads were distributed as described in the Containment Design Report. The response curve for 0.15g ground acceleration with 5-percent critical damping was used. A one-third stress increase was permitted on working stress allowables when considering the effect of seismic loads.

1.11.4.12 Shield Wall

The peak of the response curve for 0.15g ground acceleration with 5-percent critical damping was used. The pipe break loads controlled the design.

1.11.4.13 Retaining Wall At Equipment Entrance

The wall was designed for soil pressure including a 1000 psf surcharge applied during reactor loading. The load combination that includes a seismic factor governs the design. It has been shown that the wall design was adequate.

1.11.4.14 Primary Water Storage Tank and Refueling Water Storage Tank Foundation

The seismic loads on the circular wall and center pier were those supplied by the tank manufacturer. The shear force from the earthquake on the water in the tank was applied at 3/4 L above the top slab. The shear force from the earthquake on the tank was applied at L/2 above the top slab, where L = the height of the tank. The horizontal shear force from the earthquake effect on the dead weight of the foundation was determined by using the peak of the response curve for 0.15g ground acceleration with 5-percent critical damping. A triangular distribution was used. The earthquake effect of the backfill was also considered. The load was applied to the walls as the resultant of a triangular pressure distribution. The stresses were limited to working stress design limits. The temperature steel considerations controlled the design of the walls and center pier.

1.11.4.15 Condensate Water Storage Tank Foundation

The seismic loads on the spread footing foundation were those supplied by the tank manufacturer. The shear forces from the earthquake on the water in the tank were applied at 3/4 L above the footing, where L = the height of the tank. The shear force from the earthquake on the tank was applied at L/2 above the top of the footing. The stresses were limited to working stress design limits.

A multi degree-of-freedom modal analysis was performed on all Class I structures for Indian Point Unit 3. The results indicated that all structures except the containment structure were rigid. The only significant differences between the structural design of Units 2 and 3 seismic Class I buildings are the control building and the steel structural portion of the primary auxiliary building for Indian Point Unit 2, which are flexible steel structures. On Unit 3 they are rigid concrete structures. All seismic Class I structures on Indian Point Unit 2 except control building and containment shell are rigid and move with zero period ground acceleration. However, the design of all seismic Class I structures on Unit 2 were standardized and based on the peak acceleration of the ground response spectrum, which is extremely conservative for rigid structures.

In the preceding designs, limits have been placed on stresses to ensure that all structures will respond elastically to the earthquake. If for some reason inelastic response were to occur, the period of the structure would be expected to increase. Since the majority of the structures were designed for the peak of the response curve, the effect of any change in period would be to decrease the coefficient of spectral acceleration and thus lower all shears and moments.

Mathematical models were not used for seismic design of instrumentation. Ability to withstand the seismic condition is determined by actual vibration type testing of typical instrumentation equipment under simulated seismic accelerations to demonstrate its ability to perform its functions. The seismic testing is reported in Westinghouse report WCAP-7397-L (Reference 1).

The locations of protection and safeguards control and electrical equipment in Indian Point Unit 2 have been identified. The most adverse location, seismically, is the control building floor at elevation 53-ft, which supports the nuclear instrumentation, radiation monitoring, process instrumentation, and safeguards logic racks. Dynamic analyses of this building for the plant design basis earthquake of 0.15g show that the significant horizontal and vertical accelerations of this floor are within the specified low seismic test envelopes given in WCAP-7397-L (Reference 1).

Seismic analysis of Class I equipment including heat exchangers, pumps, tanks, valves, motors, and electrical equipment components was performed using one of the following four methods:

1. Equipment, which is rigid and rigidly attached to its support structure, was analyzed for a "g" loading equal to the peak acceleration of the supporting structure at the appropriate elevation.

2. Equipment, which is not rigid and therefore a potential for response to the support motion exists, was analyzed for the peak of the floor response curve for appropriate damping values.

3. In some instances nonrigid equipment was analyzed using a multi degree-of-freedom modal analysis. All contributing modes were considered. In addition, a sufficient number of masses was included in the mathematical models to ensure that coupling effects of members within the component were properly considered. The results of these analyses indicated that the models contained more masses than necessary, and that future analyses of comparable equipment could be considerably simplified by considering fewer masses. The method of dynamic analysis used a proprietary computer code called WESTDYN. This code used as input the inertia values, member sectional properties, elastic characteristics, support and restraint data characteristics, and the appropriate seismic response spectrum. Both horizontal and vertical components of the seismic response spectrum were applied simultaneously. The modal participation factors were combined with the mode shapes and the appropriate seismic response spectra to obtain the structural response for each mode. The internal forces and moments were computed for each mode from which the modal stresses are determined. The stresses were then summed using the root mean square method.

4. Type testing of selected electrical equipment has been conducted to demonstrate seismic design adequacy as described in WCAP-7397-L (Reference 1).

For the analysis of equipment to resist the vertical seismic component, two-thirds of the horizontal response spectrum curves were used to determine the acceleration appropriate to the vertical frequency.

Engineered safeguards tanks, e.g., boric acid, accumulator and surge tanks, were analyzed using method 3 above for combined horizontal and vertical seismic excitation occurring simultaneously and in conjunction with normal loads without exceeding allowable stresses. Hydrodynamic analyses of these tanks have been performed using the methods described in Chapter 6 of the "U.S. Atomic Energy Commission - TID 7024." The stresses for these components due to the above-mentioned load combinations were found to be within allowable limits. Heat exchangers associated with the engineered safeguard systems, e.g., component cooling and residual heat removal, were analyzed using method 3 above, and the results show that stresses and deflections are within allowable limits.

Selected critical engineered safeguards valves were analyzed using method 3 above and the results indicated that their fundamental natural frequency was sufficiently separated from the building frequency. The results further indicated that the total stress, considering all modes, was far below the allowable stress limit.

Appendages, such as motors attached to motor-operated valves, were included in the mathematical models.

1.11.4.16 Class I Piping Systems

Class I piping systems were designed and analyzed as described in the succeeding paragraphs. However, in an attempt to correlate the simplified method of analysis suggested by the AEC for the H. B. Robinson Nuclear Generating Station, the following discussion is presented:

If no dynamic analysis is performed on Class I piping systems, these systems for H. B. Robinson plant were to be checked to determine whether the results conform to the following formula:

$$1.3^{*}\ K\ S_s + S_n \leq 1.8\ S_a$$

[***Note*** *- *The 1.3 factor was recommended by the AEC to represent the contributions of higher modes above the fundamental mode. Detailed dynamic analyses performed on Indian Point Unit 2, and described later, indicate that where significant stresses exist in piping systems, a more realistic modal contribution factor would be 1.1. However, for the present discussion we will adhere to the 1.3 factor for additional conservatism.*]

where:

S_s - represents seismic stress including effects of valve motors, from design calculations

S_n - represents normal primary and bending stresses for loadings other than seismic, from design calculations

$1.8\ S_a$ - equals 1.8 times the allowable stress or yield stress, whichever is higher for code listed materials.

K - ratio of peak acceleration of floor response spectra to acceleration used in the piping design

The piping design criteria limited the deadweight and seismic stresses to $0.2\ S_a$. The longitudinal pressure stress is $0.5\ S_a$.

$$1.3\ K\ (0.2\ S_a) + 0.5\ S_a \leq 1.8\ S_a$$

Solving, the K-factor becomes:

K = 5

This factor combined with the 1.3 modal contribution factor gives a combined factor of 6.5, which is more than double the original suggested multiplier of 3.

Indian Point Unit 2 conservatively meets the criteria suggested for application on the H. B. Robinson Plant for seismic Class I piping.

However, a different and more detailed method of analysis was actually undertaken to illustrate the conservatism of design approach used for Indian Point Unit 2. This approach is described in detail below:

It is obviously necessary to use simplifying assumptions when performing initial design of piping systems, including restraints, rather than a dynamic analysis involving a trial and error procedure. Simplified design procedures are not uncommon and often suggested in codes, i.e., USAS B31.1 - Power Piping Code.

A complete flexibility analysis involving detailed modeling of Class I piping systems is unnecessary if the conservatism of the simplifying assumptions used in the initial design can be demonstrated. A "third party" review was conducted to establish the adequacy and conservatism of the original design criteria for Class I piping systems as performed by the architect/engineer (United Engineers and Constructors, Inc.) and the seismic restraint supplier (Bergen-Paterson Pipe Support Corp.). The review involved the following steps:

1. Representatives from Westinghouse and United Engineers and Constructors, Inc., visited the Indian Point Unit 2 site and inspected the Class I piping systems.

2. Based upon their best engineering judgment, representative worst-case lines were selected for detailed dynamic analyses.

3. In exercising their engineering judgment, these representatives looked for the following characteristics, which would indicate possible sources of problems.

    a. Amplification due to the location and elevation in building.

    b. Large concentrated masses such as overhung motor-operated valves, particularly in what appear to be flexible sections of the pipe.

    c. Complexity of configuration of the piping system itself such that application of the original design criteria would be difficult.

    d. Manual excitation of the pipe by pushing or kicking indicated excessive flexibility either in the pipe excited or the piping attached to it.

4. The results of the dynamic analyses were compared with original design values to determine whether the design approach was conservative. Besides analysis of the reactor coolant loop, portions of the following systems were analyzed:

    a. Safety injection.

b. Main steam.
c. Residual heat removal.
d. Service water.
e. Accumulator discharge.
f. Containment spray.
g. Component cooling.

1.11.4.16.1 Design Approach

The design and placement of seismic restraints were predicated on the principle of containing the seismic stresses without restricting the free thermal expansion of the piping system. The systems were designed to have sufficient flexibility to prevent the movements from causing failure of piping or anchors from overstress.

Two fundamental principles underlie the design approach, namely:

1. The system be designed such that its fundamental natural frequency does not coincide with the exciting frequency.

2. The maximum seismic stresses in piping be less than the USAS B31.1 code allowable value. The seismic stresses were limited to 0.2 S allowable (3000 psi). This is extremely conservative since the longitudinal pressure stress accounts for approximately 0.5 S allowable leaving a margin of safety of 0.5 S allowable, which is unused. (Note-this is based on a maximum allowable of 1.2 S_a)

These fundamental principles should ensure that stresses will be within code allowable stress limits, and that the piping will not go into resonance with the exciting frequency. Tables of recommended maximum spacing of supports, for straight runs of pipe, were developed. The recommended spacing of supports was modified near bends and concentrated masses (i.e. valves) to account for additional weight and flexibility.

1.11.4.16.2 Analysis Approach

In order to determine whether the design procedure resulted in an acceptable system, selected worst case Class I piping systems were modeled and a dynamic flexibility analysis performed. A detailed description of the method of analysis is given below.

The analysis was performed using a proprietary computer code called WESTDYN. The code uses as input system geometry, inertia values, member sectional properties, elastic characteristics, support and restraint data characteristics, and the appropriate Indian Point seismic floor response spectrum for 0.5-percent critical damping. Both horizontal and vertical components of the seismic response spectrum are applied simultaneously.

With this input data, the overall stiffness matrix of the three-dimensional piping system is generated (including translational and rotational stiffness's). The modal participation factors are computed and combined with the mode shapes and the appropriate seismic response spectra to give the structural response for each mode.

Each piping run is modeled as a three-dimensional system, which consists of straight segments, curved segments, and restraints. Straight segments are distinguished from curved segments during data output.

The computer code requires that the piping be represented by a discrete mass model. Each mass includes the contribution of both the steel encasement and conveyed fluid. Where valves or other concentrated masses exist in the piping system, they were included in the model.

Restraints were included in the model at their proper location. The directionality of the restraints was also considered. The detailed dynamic analyses of selected worst case Class I piping indicated that the method used to design the seismic restraints was conservative. Based on this critical review of the selected worst case systems and the consistent application of the same design procedure to all completely engineered seismic Class I systems, the seismic design of other Class I systems, not analyzed, was deemed adequate.

The maximum stresses imposed by the normal loads plus loads associated with the design-basis earthquake (DBE) are below 1.2S, where S is the allowable stress limit obtained from the Power Piping Code - USAS B31.1.0 - 1955.

Some of the items of conservatism employed in the seismic design of Class I piping systems for Indian Point Unit 2 were:

1. The maximum longitudinal stress due to seismic excitation was limited to 0.2S rather than the usual 0.7S.

2. The maximum allowable stress was limited to 1.2S. If the combination of normal and DBE loads were considered as a faulted condition, the allowable membrane and bending stresses could be chosen as those corresponding to 20-percent to 40-percent of the material uniform strain at temperature, respectively. This would give more than a factor of 2 margin between the allowable and the maximum actual stresses.

3. A low value of the fraction of critical damping was adopted (0.5-percent). Dr. N. M. Newmark recommended a value of 2-percent for vital piping at or just below the yield point. This would reduce the maximum amplification of the ground acceleration.

4. The maximum longitudinal stresses due to pressure, deadweight, and seismic loads were presumed to occur at the same cross-section and some point in the cross-section.

Some averaging of the response spectra was performed to smooth out the erratic response of the earthquake's random behavior. At the high frequency end of the spectra, the acceleration levels of the smoothed spectra converge to the values of the unsmoothed spectra.

It is therefore concluded that the design procedure used to design seismic Class I restraints for Indian Point Unit 2 is conservative.

NRC IE Bulletin (IEB) No. 79-07 was concerned with inadequacies identified in the seismic analysis of certain piping systems at several power reactors. The inadequate treatment of piping loads from earthquakes was attributed to the fact that some piping analysis codes used an algebraic summation of the loads predicted separately by computer code for both the horizontal components and the vertical component of seismic events. In accordance with the IEB, such co-directional loads should not be algebraically added unless certain more complex time-history analyses are performed. The IEB emphasized that to properly account for the effects of earthquakes on systems important to safety, such loads should be combined absolutely or by using techniques such as the sum of the squares.

In response to IE Bulletin No. 79-07, eight (8) Indian Point Unit No. 2 lines were reanalyzed using the UE&C-ADLPIPE-2 dynamic seismic computer code. This code utilizes the worst-case two-dimensional evaluation technique and uses the square root of the sum of the squares option for combining both intramodal and intermodal responses.

The difference between the newly calculated total pipe stress and the originally calculated total pipe stress is not significant. Even after applying a 1.3 "adjustment" factor to the calculated seismic stress component, the total pipe stress remains below the allowable stress limit.

Furthermore, the loads on the pipe supports and equipment nozzles were re-evaluated on the basis of the confirmatory reanalysis and found to be acceptable, as documented in Reference 9.

1.11.4.17 Reactor Coolant System Analysis for Combination Loading of Design-Basis Earthquake and Design-Basis Accident [Historical Information Only]

The Indian Point Unit 2 reactor coolant system was not committed to be designed for the combination of the seismic and blowdown loads. However, an analysis for this combination of loadings was performed for the original configuration of the Indian Point Unit 3 reactor coolant system, which was identical to the original Unit 2 configuration.

The analysis was performed as outlined below:

1. A lumped mass dynamic mathematical model of the primary coolant loop and support system was developed.

2. This dynamic model was subjected to multiple simultaneous time history hydraulic forcing functions for the blowdown analysis. The double-ended ruptures were located at places of large change in flexibility. Time history response of the total structure to these conditions was computed and reduced to time history stresses.

3. The dynamic model was then subjected to a floor response spectra earthquake analysis.

4. The loads as determined above were used for an evaluation of the stresses along the piping system.

5. The stresses as determined from the basis described above were lower than the allowable stresses calculated by using the approach described in WCAP-5890, Revision 1 (Reference 2) and the following parameters:

a. 20-percent of the uniform strain on the allowable membrane and average strain.

b. 23,100 psi as the at-temperature yield in the axial direction. This value was based on the minimum value of the at-temperature yield in the loop direction as measured with samples from the Unit 2 piping, and increased by 10-percent for the increase in strength in going from the loop to the axial direction. The tensile tests on the Unit 2 piping material at-temperature yielded at a minimum value of 20,900 psi, a maximum of 29,700 psi, and an average of eleven samples of 23,300 psi.

Based on the above analysis, it was concluded that the Unit 2 reactor coolant system can stand the combination of blowdown and seismic loads within acceptable stress limits.

In 1989, the NRC approved changes to the design bases with respect to dynamic effects of postulated primary loop pipe ruptures, as discussed in Section 4.1.2.4. In 2000, an analysis of the Unit 2 reactor coolant loop and its component supports, which incorporates the NRC approved changes, was performed with the replacement steam generators and sixteen of the original twenty-four steam generator support frame hydraulic snubbers removed (Reference 11). In line with the older analysis for Unit 3 described above, the Unit 2 analysis of 2000 included the effects of the now controlling feedwater line break at the steam generator nozzle in a similar fashion as described for the Unit 3 analysis. Based on this revised analysis, it was concluded that the Unit 2 reactor coolant system can withstand the combination of blowdown and seismic loads within acceptable stress limits.

This 2000 analysis has since been updated to include a power uprate to a core power level of 3216 MWt. Combination of blowdown and seismic loads were not considered in this latest evaluation.

1.11.4.18 Service Water Lines

The service water lines consist of two 24-in. diameter carbon steel pipes. They run in a common trench, which is backfilled. Assuming that the ends of a pipe are free to displace vertically but not rotate and that the maximum permissible stress is restricted to 30,000 psi, a parametric study showed that the following maximum allowable relative displacements may occur during a seismic disturbance without overstressing the pipe:

Length, ft	1	10	25	50	75	100
Displacement, in.	0.002	0.20	1.25	5.01	11.27	20.04

It is therefore concluded that the service water lines could withstand, without being overstressed, relative bedrock displacements associated with the earthquakes defined for the Indian Point site.

1.11.4.19 Seismic Evaluation of the Fan Cooler and Passive Hydrogen Recombiner Systems

The seismic analysis of the fan cooler system was completed in two parts.

1. Analysis of the structural steel enclosure of the fan cooler units to include the effect of supported equipment.

   The structural analysis considering simultaneous incident pressures and earthquake forces was conducted on particular members, plates, and connections, which are, considered critical to the structural performance of the reactor containment fan coolers. This analysis included consideration of the mass of all components supported partially or wholly by the enclosure. The fan, fan motor, and fan motor heat exchanger, although entirely within the enclosure, are independently supported from the concrete floor that makes up the base of each unit.

   Earthquake loadings were treated using the response spectra techniques. A horizontal force of 0.6W and a vertical force of 0.4W, where W is the weight of the member including static forces, were concentrated at the center of gravity of each member. A negative differential pressure of 1.5 psig was applied to the portion of the unit from the inlet up through the fan compartment; a negative differential pressure of 6.3 psig was applied to the charcoal filter compartment. Both of these values are consistent with unit geometry and containment environment following a loss of coolant accident or main steam line break. The charcoal filter compartment pressure is limited by a pressure equalization device installed during the 1997/1998 Maintenance Outage. All loadings were assumed to act simultaneously and comparison was made with allowable stresses consistent with specifications for installed materials. An increase in allowables was considered for loads associated with accident conditions. Where applicable, allowable concrete stresses were taken from ACI Specifications.

   Results of the analysis on the enclosure demonstrated that the design is adequate.

2. Evaluation of the fan motor system and its foundation.

   The fan motor and its supporting structural system was evaluated using acceleration values for a maximum hypothetical earthquake. These values are 0.6g for the horizontal direction and 0.4g for the vertical direction. These accelerations were assumed to act simultaneously.

   The failure modes considered for the motor unit were excess deflection of the rotor shaft, which results in rubbing against the housing or by bearing failure. The failure modes for the fan are failure of the fan shaft support bearings or deflection of the fan housing and fan wheel causing binding. In addition, the potential for shear and overturning failure of the motor fan assembly at the foundation anchorage was evaluated.

   Based on analyses made on similar fan motor systems, it was concluded the fan cooler units in the containment are adequately designed to resist the seismic loading defined for the site and supporting building structure.

The two hydrogen recombiners are located in the containment at the 95-ft elevation. The hydrogen recombiners are as shown on Figure 6.8-1. The Passive Hydrogen Recombiners (PHRs) are seismic class I, and have undergone qualification testing in accordance with IEEE 344-1987. On April 14, 2005, NRC issued IP2 License Amendment 243 which eliminated the requirement for hydrogen recombiners to provide any combustible gas control function.

1.11.4.20 Masonry Walls

In response to IE Bulletin 80-11, safety related masonry walls were evaluated to demonstrate the ability to withstand the specified design load conditions without impairment of wall integrity or the performance of required safety functions. NRC acceptance of this evaluation is documented in reference 10. As a result of this evaluation, certain walls in the control building, the Unit No. 1 Superheater building, the boric acid evaporator building, the fan house, and the fuel storage building have been reinforced.

1.11.5 Wind Effects

The IP2 licensing basis does not include tornado protection for the design of the buildings, structures and components. Tornado protection is not a design criterion for IP2. However, the following structures were evaluated for tornado loads: containment building, primary auxiliary building, control building, fuel storage building (including the spent fuel pit), and the intake structure.

Detailed information on the containment structure is found in Appendix B of the Containment Design Report. The containment structure will not be penetrated by a 4-in. x 12-in. x 12-ft wood plank traveling at 300 mph, or by a 4000-pound auto traveling at 50 mph less than 25-ft above the ground.

With respect to the primary auxiliary building, control building, and fuel storage building, information from the siding manufacturer indicates that siding panels will blow out at 170 psf, which is equivalent to a 1.18 psi negative pressure. Panels fail at 60 psf external pressure, which is equivalent to a 162 mph external wind load (60 psf controls the external loading condition). The grits will fail at 90 psf, which is equivalent to a 0.62 psi negative pressure. The 3.25-in. thick siding panels are not capable of resisting any tornado-generated missiles.

Spent fuel pit tornado protection is discussed in proprietary WCAP-7313-L. The intake structure is capable of resisting any wind or missile loads generated by a tornado. This is true for the structure itself, but does not necessarily include associated equipment.

1.11.6 Structural Effects

The potential for damage to Class I structures due to failure of nearby Class II or Class III structures, or due to failure of Class III cranes, has been considered.

The only Class I structures and components that could be endangered by failure of Class III structures are the control building, main steam piping, and feedwater piping, which could be endangered by failure of the Class III turbine building. No special provisions were provided in the original plant design except in the case of the main steam and feedwater lines up to the isolation valves, which are protected by the shield wall and the structural frame at the north end of the shield wall. Evaluations were performed and bracing was added to the turbine building during construction, as described in section 1.11.6.5, to preclude such catastrophic failures.

The only Class III crane whose failure could endanger any Class I function is the fuel storage building crane. Failure of this crane will not impair a safe and orderly shutdown. The wheels of the bridge and the trolley are shaped such that sliding perpendicular to the rail would not be possible. The lateral load from an earthquake on the trolley crane rail is about 50-percent greater than the lateral loads from impact specified by the AISC Code for design within working stress limits. The stresses on the crane rail are low due to the earthquake load. For this reason no failure of the crane rail is anticipated.

The Class III manipulator crane in the containment building is restrained from overturning and will not endanger Class I structures.

The turbine building and the fuel handling building are functionally Class III structures. However, these structures have been analyzed using a multidegree of freedom modal dynamic analysis method to ensure that there is no potential for gross structural collapse of these structures as a result of the maximum hypothetical earthquake. The results of the analyses are given below. A value of 7-percent structural damping was assumed in the analysis. Total response of the structure was determined on the basis of the square root sum of the squares basis of each mode contribution. A similar dynamic analysis was also performed to ensure that no potential gross failure of the Indian Point Unit 1 stack or superheater building could occur for the maximum hypothetical earthquake, or for the design-basis tornado for Indian Point Unit 2. The resultant dead, live, and seismic design stresses in the basic building structure is limited to 0.9 yield of the steel.

The results of specific analyses are discussed in the following sections.

1.11.6.1 Seismic Analysis of the Indian Point Unit 2 Turbine Building

A spectrum response analysis was performed for the turbine building considering the design-basis earthquake (DBE), which has a peak horizontal ground acceleration of 0.15g. The associated earthquake response spectrum is shown in Figure 1.11-2.

The foundation was considered rigid since the footings for the structural frames of the building are underlaid by either rock or a lean concrete, which bears on rock. Also, in the analysis, interaction between the turbine and the structural frame for the building was neglected. The analysis, as performed, represents a linear elastic system.

The analysis of the turbine building was performed under the assumption that the north-south motions, east-west motions and vertical motions will be uncoupled. The dynamic analysis effort was limited only to horizontal motions in the east-west and north-south directions. However, vertical components of the earthquake were considered by adding a 0.13g component to dead loads. Each of the models was simulated for the computer program called STARDYNE. A description of the modeling capabilities of STARDYNE are contained in "STARDYNE Structural Analyses Systems Users' Manual" prepared by Mechanics Research, Inc., for Control Data Corporation.

The STARDYNE program was used in three ways. First, the portal frames were analyzed for a static unit force at each portal to determine their resistance to horizontal motions resulting from the turbine bay crane. This information was incorporated into the model for the analysis of the crane girder to determine the distribution of horizontal turbine bay crane loads to the various east-west portal frames. Secondly, the program was used to determine the forces induced in the frames as a result of gravity forces, and, thirdly, the STARDYNE program was used to determine the fundamental frequencies of each of the models and the characteristic shapes. In addition, the STARDYNE program is also capable of determining the modal member forces for each of the fundamental frequencies. This information for each model and mode was stored on tape along with the gravity forces for each model and later used in an earthquake analysis program to determine the maximum probable deflection, acceleration, member forces, member stresses, and the combined gravity plus earthquake member stress responses. Dynamic characteristics of the turbine building are shown in Table 1.11-4.

Results of the analysis indicated that the 0.9 Fy combined load allowable stress was not violated except locally in the flange of columns where cross bracing framed in eccentric to other joint members. Reduction of stresses to allowable values is accomplished by the addition of flange cover plates.

While allowable stresses in the cross bracing did not exceed the 0.9 yield stress allowable, it was determined that most of the "x" cross bracing would buckle at very low compressive stress due to high ℓ/r ratios. In order to assure the lateral stiffness of the bents and load carrying capacity as determined in the analysis, cover plates were attached to the bracing equal to the original area of the "x" crossing bracing. This assures design adequacy with only "x" cross bracing in tension assumed to be active in carrying lateral load.

1.11.6.2 Seismic Evaluation of the Fuel Storage Building Structure Above the Spent Fuel Pit

The fuel storage building for Indian Point Unit 2 consists of the spent fuel pit constructed of reinforced concrete and founded on rock. The fundamental frequency of the pit is approximately 22 cps and therefore can be considered rigid. The steel superstructure above the pit encloses the pit and supports the fuel cask handling crane. This superstructure was designed as a Class III structure. The seismic loads used in the analysis of the steel superstructure were as follows:

1. Zero period ground acceleration: 0.15g horizontal, 0.10g vertical.
2. 7-percent damping.
3. Response spectrum curve as defined in Figure 1.11-2.
4. Inertial forces for each mass point are determined on the basis of the square root of the sum of the squares.

A dynamic multidegree of freedom, modal analysis of the structure was constructed as shown in Historical Figures 1.11-3 and 1.11-4. The stiffness properties of the elements were determined by the combined stiffness of the frame bents in the north-south and east-west directions taken separately. The stiffness of each bent was determined by the computer program STRUDL. The total inertial forces determined by the dynamic analysis were distributed to each individual bent and resultant member stresses were determined. The crane was assumed fully loaded. Evaluation of these seismic stresses show maximum stresses occurring in diagonal bracing. The maximum stress thus determined in the cross bracing was 18.5 ksi. The maximum combined dead and seismic column load stress determined by the analysis was 12.8 psi compression.

On the basis of these results it was determined that the fuel storage building superstructure was adequately designed to carry the seismic load defined for the site.

In addition to the analysis of the building structure, the fuel crane bridge was evaluated to determine the potential for the crane bridge to lift off its track support in the event of a seismic disturbance. The vertical mode fundamental frequency of the fuel storage building is approximately 9 cps.

The crane bridge has also been analyzed dynamically both loaded and unloaded and for various positions of the trolley. It was determined that the crane with the trolley at the end of the span and unloaded would have a fundamental frequency of approximately 9 cps. Considering potential resonance with the fundamental vertical mode of the building at 9 cps the resulting g-loading was 1.05g. The only potential for crane lift-off will be in the unloaded condition with the trolley parked near the support. Since the unloaded crane will not be parked over the pool no potential hazard exists and vertical restraints are not required.

1.11.6.3 Seismic and Wind Analysis of the Superheater Stack of Indian Point Unit 1

The Indian Point Unit 1 superheater stack has been analyzed for seismic, tornado, and vortex-shedding wind load effects. The results of this analysis are summarized below. As a result of this analysis on the existing stack it is concluded:

1. The stack can withstand a tornado wind load of approximately 300mph prior to buckling failure of the stack steel shell.

2. The maximum stress in the stack at the critical vortex-shedding frequency wind velocity is 7660 psi, which provided a 3.64 factor of safety against stack failure by this mode.

3. The maximum combined dead and seismic stress for the earthquake parameters defined for the site is 19,140 psi, which provides a 1.46 factor of safety against stack failure by this mode.

1.11.6.3.1 Load Case 1 - Tornado

I. Load Criteria

Wind = 300 mph
L = D + W'

where:

L = Total load
D = Dead load
W'= Tornado load

II. Method of Load Analysis

As prescribed in ASCE Paper 3269 for uniform wind velocity with height; no gust factor.

III. Allowable Stress Criteria

$$\sigma_a = \frac{0.72Et}{\pi(1 - \nu^2)r} = 27{,}900 \text{ psi}$$

where:

σ_a = allowable stress (psi)

E = modulus of elasticity (psi)

t = shell thickness (in.)

ν = Poisson's ratio

r = radius of stack (in.)

IV. Stress Determination

$$\sigma = \frac{D}{A} + \frac{W'\bar{y}r}{I} = 1.54 + 25.75 = 27.29 \text{ ksi}$$

where:

$\bar{y}$ = centroidal height of stack (in.)

I = moment of inertia of stack (in.^{4})

A = cross sectional area of stack (in.^{2})

$$\text{Factor of Safety} = \frac{\sigma_a}{\sigma} = \frac{27.9}{27.29} = 1.02$$

1.11.6.3.2 Load Case 2 - Seismic

I. Load Criteria

a) Zero period ground acceleration: 0.15 g horizontal; 0.10 g vertical.

b) Damping 7-percent.

c) Ground response curve - Figure 1.11-2.

$$L = D + E'_{h'} = E'_V$$

where:

E'_h = load resulting from horizontal earthquake component

E'_V =load resulting from vertical earthquake component

II. Method of Load Analysis

Multidegree of freedom modal analysis of the superheater building and stack as shown in Figure 1.11-5. The square root of the sum of the squares of seismic inertia forces at mass points is used to determine resultant shears and moments in the stack.

III. Allowable Stress Criteria

See Load Case 1, item III.

IV. Stress Determination

$$\sigma = \frac{D}{A} + \frac{E'v}{A} + \frac{E_h \overline{Xr}}{I}$$

σ= 1.54 + 0.20 + 17.4 = 19.14

Factor of Safety = $\frac{\sigma_a}{\sigma} = \frac{27.9}{19.14}$

= 1.46

where:

$\overline{X}$ = lever arm of node inertia force

1.11.6.3.3 Load Case 3 - Vortex-Shedding

I. Expression for maximum uniformly distributed force due to vortex-shedding.

$$P = (MF) 1/2 \rho v^2 \times C_L \times D \times L \frac{\pi}{\delta}$$

C_L = Lift coefficient for a stationary circular cylinder

MF = A multiplying factor applied to the lift coefficient to account for a vibrating cylinder

D = Average stack diameter (ft)

L = Length of stack (ft)

δ = Logarithmic decrement

ρ = Air density (0.0023385 lb - sec^2/ft^4)

v = F1 x Vc

Vc = Critical vortex-shedding velocity (fps)

F1 = A correction factor, which accounts for the fact that stack oscillations have occurred as high as 30-percent above shedding velocity

$$Vc = \frac{f \times D}{S}$$

S = Stronhal number

f = Fundamental frequency (cps)

II. Pertinent parameters

CL = 0.1

MF = 4.0

D = 20-ft

L = 334.5-ft

$\delta = 0.04\pi$ (2-percent critical damping)

Vc = 42.7 fps

F1 = 1.2

S = 0.27

f = 0.576 cps

III. Stress criteria

$$\sigma = \frac{D}{A} + \frac{Phr}{2I} = 1.54 + 6.12 = 7.66 \text{ksi}$$

Factor of Safety = $\frac{\sigma_a}{\sigma} = \frac{27.9}{7.66} = 3.64$

In addition to the analysis performed for the existing stack it was determined that the stack with 80-ft removed from the top would have the capacity to resist a 360 mph wind for the criteria as defined in Load Case I; the seismic as defined in Load Case II; and the vortex-shedding as defined in Load Case III.

1.11.6.4 Seismic and Tornado Evaluation of the Superheater Building at Indian Point Unit 1

A spectrum response analysis was performed for the superheater building considering the design basis earthquake, which has a maximum horizontal ground motion of 0.15g. A dampening coefficient equal to seven percent was assumed for all modes. The earthquake response spectra used is shown in Figure 1.11-2 normalized to 0.15g zero period ground acceleration. In the analysis no interaction with the foundation was considered since the footings for the structural frame for the building are underlaid by rock. Also, in the analysis, the stiffness interaction between the turbine building and the structural frame for the superheater building was neglected, but the mass of the turbine building was included in the dynamic analysis. The analysis, as performed, represents a linear elastic system.

The analysis of the superheater building was performed under the assumption that the north-south motions, east-west motions, and vertical motions were uncoupled. The analysis effort was limited only to horizontal motions in the east-west and north-south directions, and no attempt was made to model vertical motions or to combine vertical and horizontal motions. However, vertical seismic motions have been considered in the results by increasing the dead load stress in building members by a factor equal to two thirds of the combined mode horizontal inertial g-load as determined in either the east-west or north-south direction.

In each direction, north-south and east-west, the column lines were modeled in detail. These structural models were developed for elastic-static analyses obtained from the computer program STRUDL. They were used for two purposes: to develop the master stiffness matrices associated with the two directions, east-west and north-south, used in the dynamic analyses; and to determine resultant member stresses using the equivalent static seismic forces determined from the dynamic analyses.

The dynamic characteristics, frequencies, and mode shapes of the superheater building were determined using the Westinghouse computer program SAND. The equivalent static forces resulting from the dynamic response were developed using a response spectrum seismic analysis performed by the Westinghouse computer program SPECTA.

The equivalent static force associated with a particular mass resulting from a dynamic response is defined as the square root of the sum of the squares of the equivalent static forces associated with that mass for each mode. The equivalent static force associated with a mode and a mass point is defined as the value of the mass times the maximum acceleration associated with the mass point for that particular mode. The maximum acceleration associated with a mode and mass point is defined as follows:

$$(\ddot{U}_{rn})\,\text{Max} = (\ddot{A}_n)\,\text{Max}\,\varnothing_{rn}$$

$$(\ddot{A}_n)\ Max = \Gamma_n\ Sa_n$$

$$\Gamma_n = \frac{\sum_r M_r \varnothing'_{rn}}{\sum_r M_r \varnothing^2_{rn}}$$

Where:

n = Refers to mode n

r = Refers to mass r

$\varnothing'_{rn}$ = Component of $\varnothing_{rn}$ in the direction of the earthquake

$\varnothing_{rn}$ = Component of mode shape n for mass r

M_r = Mass lumped at point r

$(\ddot{A}_n)Max$= Maximum modal acceleration for mode n

Sa_n= Spectral acceleration for mode n from response curve for 7-percent damping

$(\ddot{U}_{rn})Max$ = Maximum acceleration in mode n for mass point r

Γ_n = Modal participation factor for mode n

Sectional views in the north-south and east-west directions are shown in Figures 1.11-5 and 1.11-6. A typical column line modeled for STRUDL to determine overall column line stiffness and permit determination of resultant seismic stresses is shown in Figure 1.11-7. In Figure 1.11-8 is presented the dynamic model used to determine inertial forces.

Results of the analysis showed several column lines contained diagonal bracing with stresses, which exceeded the allowable stress value of 0.9 f_y. In addition several of the cross bracings showed compressive stress levels, which exceeded the expected buckling stress as determined by the ℓ/r ratio for the member. Overstressed members can be strengthened by attaching cover plates to the angle bracing. In a few instances columns were found to be locally overstressed due to eccentric positioning of cross bracing. These areas can be reinforced by flange cover plates. Approximately 30 tons of additional plate will strengthen the structure.

With respect to tornado resistance of the structure, total lateral load in the north-south direction is approximately 10-percent, and in the east-west direction 20-percent, less than the seismic-induced lateral load on the structure.

Tornado loads were based on a 360-mph wind using the shape factors for a rectangular building as defined in ASCE Paper 3269. It was assumed that 20-percent of the wall area of the building was still intact as a reaction surface for the wind in addition to the total surface area of major equipment and the stack at its existing height. On the basis of this analysis, the building has approximately the same resistance capacity to a 360-mph tornado wind as it does for the 0.15g earthquake.

1.11.6.5 Evaluation of Structural Modifications

In the analysis of the superheater and turbine buildings under lateral loads, the following connections were examined:

1. Gusset plates.
2. Check of connections between beams and columns to determine their adequacy to transfer horizontal shear load.
3. Check of connections at column bases in the foundation to determine their ability to transfer the given horizontal shear load. For those column base connections subjected to a net uplift load, an analysis has been performed to ensure that they are adequate for these loads.

If it was found that a connection was inadequate to support the given load, it was redesigned.

It is not necessary to reanalyze the turbine building after the redesign because the building stiffness characteristics are essentially the same as those assumed in the initial analysis. This is because the significant fixes involved the cross bracing system, which is made up of pairs of cross bracing members. In the initial analysis, both sets of cross bracing were assumed active. However, the bracing system was such that cross members would buckle under a very small compressive load. Therefore, lateral building load must be carried in tension by the bracing system.

The fix used in the redesign was to double the area of cross bracing. The bracing in compression, due to buckling, is not active in resisting lateral building load. Therefore, only half of the cross bracing assumed in the initial analysis, which is in tension, resists this load. However, since the area of cross bracing has been doubled, the resultant effective lateral resistance is the same as that assumed in the original analysis.

An initial analysis was made of the superheater building using the existing design parameters. After completion of the analysis, the overstressed members were strengthened and a dynamic reanalysis made.

Tables 1.11-5, 1.11-6, and 1.11-7 give the relative comparisons in stiffness, horizontal inertial load, and frequency between the initial analysis and the reanalysis.

Subsequently, retired Unit 1 superheater-associated equipment has been removed from certain areas of the superheater building and the areas refurbished to provide permanent administrative facilities. These areas do not contain any safety-related equipment. The total loading on the superheater building has been reduced from the original design loading due to the removal of superheater-associated equipment. Therefore, the administrative facilities will not adversely affect the response of the superheater building during a safe-shutdown earthquake.

1.11.7 Seismic Qualification For Safe Shutdown

In response to NRC Generic Letter (GL) 87-02 and Supplement No. 1 to GL 87-02, "Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue A-46," Con Edison committed to implement Generic Implementation Procedure (GIP-2) including the clarifications, interpretations, and exceptions in the NRC's Supplemental Safety Evaluation Report (SSER-2). The NRC accepted Consolidated Edison's response and commitments regarding this issue as documented in their Safety Evaluation Report (SER) dated November 19, 1992. Consolidated Edison has verified the seismic capabilities of equipment required for safe shutdown, as documented in Reference 7. The verification utilized the Generic Implementation Procedure developed by the Seismic Qualification Utility Group and approved by the NRC. A Summary Report, including the Seismic Evaluation and Relay Evaluation Reports, was submitted to the NRC on December 31, 1996[7]. Indian Point 2 site specific SER, dated November 8, 2000, Reference 15, provides NRC conclusions that the licensee may revise its licensing basis by incorporating the SQUG methodology.

Revision 3 of the Generic Implementation Procedure (GIP-3), Reference 12, as modified and supplemented by the NRC's Supplemental Safety Evaluation Report No.2 (SSER-2), Reference 13, and No.3 (SSER-3), Reference 14, may be used as an alternative to existing methods for the seismic design and verification of modified, new, and replacement equipment. Only those portions of GIP-3 as described in Section 5 of "Implementation Guidelines for Seismic Qualification of New and Replacement Equipment/Parts (NARE) using the Generic Implementation Procedure", Reference 16, shall apply to the seismic design and verification of mechanical and electrical equipment, electrical relays, tanks and heat exchangers, and cable and conduit raceway systems.

1.11.8 Protection from Flooding of Equipment Important to Safety

In response to NRC Guidelines for Protection from Flooding of Equipment Important to Safety, Consolidated Edison identified the potential sources of flooding outside containment that could affect safety-related equipment. The areas containing safety-related equipment that could be subject to flooding from postulated failure of water systems that are not seismic Class I were evaluated. The plant is designed so as to minimize or eliminate the vulnerability of safety-related equipment to this flooding. Modifications were made to install water level alarm switches in the adjoining Unit 1 condenser pit area, and add flap panels in doors from the primary auxiliary building and the auxiliary feed pump room. A later modification was made to install a flood control drain line and valve from the PAB to a manhole in the transformer yard. These modifications along with the implementation of an alternate safe shutdown capability, as discussed in Section 8.3, serve to mitigate the consequences of the postulated flooding. Additionally, operator action would be taken in the event of flooding from the circulating water system to prevent damage to the 480 volt switchgear in the control building.

In their Safety Evaluation Report (SER) dated December 18, 1980, the NRC determined that design features and operating procedures provide assurance that the plant can be safely shut down in the event of flooding outside containment from a non-seismic component or pipe and that their guidelines (contained in Appendix A to the SER) have been satisfied.[8] The Fire Protection piping added later in the PAB was analyzed for seismic loads and restrained to prevent breakage.

REFERENCES FOR SECTION 1.11

1. E. L. Vogeding, Topical Report - Seismic Testing of Electrical and Control Equipment, WCAP-7397-L, Westinghouse Electric Corporation, January 1970.

2. Westinghouse Electric Corporation, Ultimate Strength Criteria to Ensure No Loss of Function of Piping and Vessels Under Earthquake Loading, WCAP-5890, Revision 1.

3. NRC Generic Letter, Relaxation in Arbitrary Intermediate Pipe Rupture Requirements, G.L. 87-11, dated June 19, 1987.

4. NRC Branch Technical Position MEB 3-1, Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment.

5. Letter (Attachment B) from S. Bram, Con Edison, to NRC, Subject: Request for License Amendment to Technical Specification Modifying Spent Fuel Storage Requirements, dated June 20, 1989.

6. Letter (Attachment I) from S. Bram, Con Edison, to NRC, Subject: Indian Point Unit No. 2 Spent Fuel Storage Capacity Increase, dated January 19, 1990.

7. Letter from Quinn, Con Edison, to NRC, Subject: Summary Report for Resolution of USI-A-46, Seismic Qualification, dated December 31, 1996.

8. Letter from S. A. Varga, NRC, to J. D. O'Toole, Con Edison, Subject: Safety Evaluation Report Susceptibility of Safety-Related Systems to Flooding from Failure of Non-Category I Systems, dated December 18, 1980.

9. Letter from Cahill, Con Edison, to A. Schwencer, Director of Nuclear Reactor Regulation NRC, Subject: Supplemental Response to IE Bulletins 79-02 and 79-07, dated November 27, 1979.

10. Letter from Steven A Varga, NRC to John D. O’Toole Con Edison, Subject: Completion of IE Bulletin 80-11, “Masonry Wall Design” for Indian Point Nuclear Generating Unit No. 2 (IP2), (Safety Evaluation Report included) dated October 19,1983.

11. Altran Corporation, Technical Report No. 00222-TR-001, Rev. 1, Reactor Coolant Loop Analysis for Replacement Steam Generators and Snubber Reduction

12. Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment, Revision 3, Updated 05/16/97 (GIP-3). Prepared by SQUG and sent to the NRC by letter dated May 16,1997.

13. Supplement No.1 to Generic Letter (GL) 87-02 that transmits Supplemental Safety Evaluation Report No.2 (SSER-2) on SQUG Generic Implementation

Procedure Revision 2, as Corrected on February 14, 1992 (GIP-2), May 22, 1992.

14. NRC letter to SQUG dated December 4, 1997, Supplemental Safety Evaluation Report No.3 (SSER-3), on the Review of Revision 3 to the Generic Implementation Procedure for Seismic Verification of Nuclear Power Plant Equipment, Updated 05/16/97 (GIP-3)

15. Indian Point Nuclear Generating Station No. 2 – “Plant Specific Safety Evaluation Report for Unresolved Safety Issue A-46 Program Implementation”, November 8, 2000.

16. “Implementation Guidelines for Seismic Qualification of New and Replacement Equipment/Parts (NARE) Using the Generic Implementation Procedure (GIP)”, Revision 4, by MPR Associates

TABLE 1.11-1
Damping Factors

COMPONENT	PERCENT OF CRITICAL DAMPING
Containment structure	2.0
Concrete support structure of reactor vessel	2.0
Steel assemblies:	
Bolted or riveted	2.5
welded	1.0
Vital piping systems	0.5
Concrete structures above ground	
Shear Wall	5.0
Rigid Frame	5.0

TABLE 1.11-2
Loading Combinations and Stress Limits

Loading Combinations	Vessels[1]	Piping	Supports
1. Normal loads	$P_m \leq S_m$ $P_L + P_B \leq 1.5\ S_m$	$P_m \leq S$ $P_L + P_B \leq S$	Working stresses or applicable factored load design values
2. Normal + design earthquake loads	$P_m \leq S_m$ $P_L + P_B \leq 1.5\ S_m$	$P_m \leq 1.2S$ $P_L + P_B \leq 1.2S$	1-1/3 working stresses or applicable factored load design values
3. Normal + maximum potential earthquake loads	$P_m \leq 1.2S_m$ $P_L + P_B \leq 1.2\left(1.5S_m\right)$	$P_m \leq 1.2S$ $P_L + P_B \leq 1.2\left(1.5S\right)$	Deflections and stresses of supports limited to maintain supported equipment within their stress limits
4. Normal + pipe rupture loads	$P_m \leq 1.2S_m$ $P_L + P_B \leq 1.2\left(1.5S_m\right)$	$P_m \leq 1.2S$ $P_L + P_B \leq 1.2\left(1.5S\right)$	Deflections and stresses of supports limited to maintain supported equipment within their stress limits

Where: P_m = primary general membrane stress; or stress intensity
P_L = primary local membrane stress; or stress intensity
P_B = primary bending stress; or stress intensity
S_m = stress intensity value from ASME B and PV Code, Section III
S = allowable stress from USAS B31.1 Code for Pressure Piping

notes:
1. Limited to vessels designed to ASME, Section III, Class A (or Class 1) rules. Otherwise use piping for stress limits.

TABLE 1.11-3
DELETED

TABLE 1.11-4
Dynamic Characteristics of the Turbine Building

MODE No.	Frequency (cps)	Values
1	0.5042	0.08
2	1.6141	0.12
3	2.2849	0.19
4	4.3292	0.2
5	5.2813	0.2
6	8.2814	0.18
7	12.1704	0.15
9	15.1274	0.15
10	20.754	0.15
11	22.4809	0.15
12	23.8001	0.15
13	27.3040	0.15
14	33.9678	0.15

TABLE 1.11-5
Relative Stiffness Percentages

	Percentage Increase In Stiffness Between First And Second Analysis (Percent)	
RELATIVE LOCATION IN SUPERHEATER BUILDING	EAST-WEST DIRECTION	NORTH-SOUTH DIRECTION
BOTTOM	8	56.7
MIDDLE	18.3	41.4
TOP	19.9	10.4

TABLE 1.11-6
Inertial Loads

Relative Location in Superheater Building	Inertial Loads for First and Second Analysis (Units: Kips)			
	East-West Direction		North-South Direction	
	Original	Reanalysis	Original	Reanalysis
Bottom	908	908	1091	1102
Middle	1888	1914	1687	1803
Top	1242	1271	1082	1181

TABLE 1.11-7
Frequencies

Frequencies For First And Second Analysis
(Units: cps)

	EAST-WEST DIRECTION		NORTH-SOUTH DIRECTION	
MODE	ORIGINAL	REANALYSIS	ORIGINAL	REANALYSIS
1	0.94	1.0	0.72	0.88
2	2.07	2.15	1.58	2.13
3	4.08	4.19	3.47	4.12

1.11 FIGURES

Figure No.	**Title**
Figure 1.11-1	Ten Percent of Gravity Response Spectra
Figure 1.11-2	Fifteen Percent of Gravity Response Spectra
Figure 1.11-3	Fuel Storage Building North-South Model [Historical]
Figure 1.11-4	Fuel Storage Building East-West Model [Historical]
Figure 1.11-5	Indian Point Unit 1 Superheater Building North-South Section
Figure 1.11-6	Indian Point Unit 1 Superheater Building East-West Section
Figure 1.11-7	Column Line “G”
Figure 1.11-8	Representation of Lumped Mass Model of Superheater Building Used in Dynamic

	Analysis

1.12 INSERVICE INSPECTION AND TESTING PROGRAMS

1.12.1 General

The ISI Program complies with the requirements of 10 CFR 50.55a and is based upon the requirements set forth in ASME Boiler and Pressure Vessel Code, Section XI and by the applicable Code year. This program is also responsive to pertinent provisions of applicable Regulatory Guides.

The Indian Point Unit 2 Inservice Inspection (ISI) and Testing (IST) Programs for the ten year interval are controlled by Program Plans and plant procedures.

1.12.2 Application

The ISI program applies to Quality Groups A, B, and C systems, components (including supports), and pumps and valves as classified in accordance with Regulatory Guide 1.26, Revision 3.

1.12.3 Program Summary

The ISI and IST programs identify the specific systems, components, or parts thereof to be examined and the specific pumps and valves to be tested.

1.13 CONTROL OF HEAVY LOADS

In response to a December 22, 1980 Generic Letter and to NRC Staff guidelines provided in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," Con Edison performed evaluations of provisions for the handling and control of heavy loads in the vicinity of irradiated fuel or safe shutdown equipment. Control of heavy loads in the Fuel Storage Building is addressed in section 9.5.6. The NRC documented their acceptance of Con Edison's assessments in a Safety Evaluation Report dated February 19, 1985.